Docket No. 50-271
BVY 04-058

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263

Extended Power Uprate – Supplement No. 8

Response to Request for Additional Information

REDACTED AND NON-PROPRIETARY INFORMATION

# RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATED TO EXTENDED POWER UPRATE REQUEST VERMONT YANKEE NUCLEAR POWER STATION

## PREFACE

The following information is provided in response to NRC's request for additional information dated May 28, 2004, regarding the application by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy or Vermont Yankee)) for a license amendment to increase the authorized thermal power level of the Vermont Yankee Nuclear Power Station. The individual requests for additional information (RAIs) are repeated as provided in NRC's letter of May 28.

Each of the topics identified in the NRC staff's May 28, 2004 letter has been the subject of discussions held during conference calls between the staffs of the NRC and Vermont Yankee to further clarify the information needs of the NRC staff. In certain instances the requests for additional information (RAIs) were modified based on clarifications and understandings reached during the telecons. The information provided herein is consistent with those understandings.

For convenience, a list of acronyms is included.

## PROPRIETARY INFORMATION NOTICE

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GE proprietary information is identified by a double underline inside double square brackets. In each case, a superscript notation (i.e., {3}) refers to a paragraph of the affidavit provided in Attachment 4, which documents the basis for the proprietary determination.

List of Acronyms – Extended Power Uprate

| | |
|---|---|
| AC | Alternating Current |
| ADS | Automatic Depressurization System |
| ADHR | Alternate Decay Heat Removal |
| AL | Analytical Limit |
| ANS | American Nuclear Society |
| ANSI | American National Standards Institute |
| AOO | Anticipated Operational Occurrence |
| AOV | Air Operated Valve |
| ASME | American Society of Mechanical Engineers |
| AST | Alternative Source Term |
| ATWS | Anticipated Transients Without Scram |
| BOP | Balance-of-Plant |
| BWR | Boiling Water Reactor |
| BWROG | BWR Owners Group |
| BWRVIP | BWR Vessel Internals Project |
| CDF | Core Damage Frequency |
| CFD | Computational Fluid Dynamics |
| CFR | Code of Federal Regulations |
| CLB | Current Licensing Basis |
| CLTP | Current Licensed Thermal Power |
| CLTR | CPPU Licensing Topical Report |
| COLR | Core Operating Limits Report |
| CPPU | Constant Pressure Power Uprate |
| CRTP | Current Rated Thermal Power |
| CS | Core Spray |
| CUF | Cumulative Usage Factor |
| DAS | Digital Acquisition System |
| DBA | Design Basis Accident |
| DC | Direct Current |
| DL | Dynamic Loading |
| DW | Drywell |
| EAC | Environmental Advisory Committee |
| ECCS | Emergency Core Cooling System |
| ELTR | Extended Power Uprate Licensing Topical Report |
| ENN | Entergy Nuclear Northeast |
| EOP | Emergency Operating Procedure |
| EOS | Emergency Overspeed |
| EPR | Electric Pressure Regulator |
| EPRI | Electric Power Research Institute |
| EPU | Extended Power Uprate |
| EQ | Environmental Qualification |
| ESF | Engineered Safety Feature |
| FAC | Flow-Accelerated Corrosion |
| FEA | Finite Element Analysis |
| FFT | Fast Fourier Transform |
| FIV | Flow Induced Vibration |
| FPCDS | Fuel Pool Cleanup and Demineralizer System |
| FW | Feedwater |
| G-K | Gido-Koestel |
| GDC | General Design Criterion |

| | |
|---|---|
| GE | General Electric |
| GEES | GE Energy Services |
| GENE | General Electric Nuclear Engineering |
| GL | Generic Letter |
| GNF | Global Nuclear Fuel |
| GRMS | Gravity Root Mean Square |
| HAZ | Heat Affected Zone |
| HELB | High Energy Line Break |
| HEM | Homogeneous Equilibrium Critical Flow Model |
| HEP | Human Error Probability |
| HP | High Pressure |
| HPCI | High Pressure Coolant Injection |
| HTC | Heat Transfer Coefficient |
| HVAC | Heating Ventilation and Air Conditioning |
| Hx | Heat Exchanger |
| ICF | Increased Core Flow |
| IGSCC | Intergranular Stress Corrosion Cracking |
| INPO | Institute of Nuclear Power Operations |
| IST | Inservice Testing |
| JCO | Justification for Continued Operation |
| LOCA | Loss-of-Coolant Accident |
| LP | Low Pressure |
| LPCI | Low Pressure Coolant Injection |
| LTP | Long Term Program |
| LTT | Large Transient Testing |
| MAAP | Modular Accident Analysis Program |
| MAX | Maximum |
| MDLM | Mist Diffusion Layer Model |
| MIN | Minimum |
| MOV | Motor Operated Valve |
| MPR | Mechanical Pressure Regulator |
| MS | Main Steam |
| MSL | Main Steam Line |
| MSIV | Main Steam Isolation Valve |
| MSSS | Main Steam Supply System |
| MWt | Megawatts Thermal |
| N/A | Not Applicable |
| NAI | Numerical Applications, Inc. |
| NDE | Non-Destructive Examination |
| NOS | Normal Overspeed |
| NPSH | Net Positive Suction Head |
| NRC | Nuclear Regulatory Commission |
| OFS | Orificed Fuel Support |
| OLTP | Original Licensed Thermal Power |
| OOS | Out-of-Service |
| OPL | Operating Parameter List |
| ?P | Differential Pressure |
| PCIS | Primary Containment Isolation System |
| PCPL | Primary Containment Pressure Limit |
| PI | Project Instruction |
| PRA | Probabilistic Risk Assessment |
| PSA | Probabilistic Safety Analysis |
| PUSAR | Power Uprate Safety Analysis Report |

| | |
|---|---|
| QA | Quality Assurance |
| QAP | Quality Assurance Program |
| QAPM | Quality Assurance Program Manual |
| QC2 | Quad Cities Unit 2 |
| RAI | Request for Additional Information |
| RAW | Risk Achievement Worth |
| RCIC | Reactor Core Isolation Cooling |
| RG | Regulatory Guide |
| RHR | Residual Heat Removal |
| RHRHX | Residual Heat Removal Heat Exchanger |
| RHRSW | Residual Heat Removal Service Water |
| RIPD | Reactor Internal Pressure Difference |
| RLA | Reload Licensing Analysis |
| RMS | Root Mean Square |
| RPV | Reactor Pressure Vessel |
| RPV-ED | Reactor Pressure Vessel Emergency Depressurization |
| RRU | Reactor Recirculation Unit |
| RTP | Rated Thermal Power |
| SAFDL | Specified Acceptable Fuel Design Limit |
| SBO | Station Blackout |
| SE | Safety Evaluation |
| SFP | Spent Fuel Pool |
| SGTS | Standby Gas Treatment System |
| SIL | Service Information Letter |
| SORV | Stuck Open Relief Valve |
| SQA | Software Quality Assurance |
| SRLR | Supplemental Reload Licensing Report |
| SRP | Standard Review Plan |
| SRSS | Square Root Sum of Squares |
| SRV | Safety/Relief Valve |
| SSC | System Structure Component |
| SSE | Safe Shutdown Earthquake |
| SW | Service Water |
| TEF | Top of the Enriched Fuel |
| TS | Technical Specifications |
| UFSAR | Updated Final Safety Analysis Report |
| VOQAM | Vermont Yankee Operational Quality Assurance Manual |
| VY | Vermont Yankee |
| VYNPS | Vermont Yankee Nuclear Power Station |
| WW | Wetwell |

## Materials and Chemical Engineering Branch (EMCB)

### RAI EMCB-A-1

During the Spring 2004 refueling outage at VYNPS, the licensee identified cracking in the steam dryer. The licensee has repaired some cracks and evaluated others acceptable for return to power at the current licensed thermal power (CLTP) level. The NRC staff requires the following information due to concerns that the proposed EPU conditions could cause the cracks to grow to a size that could effect the integrity of the steam dryer and could cause loose parts.

For any detected flaw in the steam dryer left unrepaired, provide a structural integrity evaluation and identify the critical flaw size for EPU conditions and the margins between the critical flaw size and the flaw size projected for the period of time that these flaws will remain inservice. The analysis should consider the potential impact on flaw growth due to the proposed EPU conditions, intergranular stress corrosion cracking (IGSCC) and fatigue. The margins should be compared to those specified in IWB-3600 of Section XI of the American Society of Mechanical Engineers (ASME) Code. An assumed IGSCC crack growth rate should be compared to those specified in NUREG-0313.

### Response to RAI EMCB-A-1

As with many previous evaluations of visual indications in drain channel locations and vane bank end plate flaws, GE developed the justification for continued operation (JCO) using qualitative, sound engineering evaluation arguments. This is an alternative to IWB-3600 ASME XI flaw growth margins analysis. These discussions are presented in the response to RAI EMCB-B-2. To address this request, each flaw has been further evaluated with the details of that evaluation presented below.

Drain Channel DC-V04C

For the case of the VYNPS dryer drain channel cracking, the inspection evaluation (discussed further in the RAI response EMCB-B-2) established that this crack is IGSCC with no evidence of fatigue extension at current licensed thermal power operating condition. It is located in the heat-affected-zone adjacent to the weld and follows the grain boundaries, exhibiting a jagged appearance. The crack is not straight and does not have any characteristics of a fatigue crack. Figure EMCB-A-1-1 schematically shows the length and location of the 12 inch IGSCC crack. The initial engineering assessment dispositioned the flaw based on qualitative factors: (1) the drain channel flaw is in the non-structural portion of the dryer (i.e., they are not located in the skirt structure itself), (2) the drain channel flaw is only 13% of the length of the weld, (3) there is a lack of any field experience of drain channel cracking extending the entire length of the weld and (4) a postulated full length crack extending the full length of the weld if postulated to propagate by fatigue during future operation, would still not generate a loose piece. Two sides of the skirt plate and the pipe attachment provide enough structure to maintain integrity of the cracked location. The field experience supports the as-is operation decision, particularly in the context that the indication will be re-inspected at the next outage. However, in order to address the request for a structural analysis, the IGSCC flaw was also assessed quantitatively.

The 12 inch long IGSCC flaw was evaluated for operation over the upcoming fuel cycle. This process required two steps. First, IGSCC crack growth was assumed during this future operation at a rate of $5x10^{-5}$ in/hr on each end, consistent with established BWRVIP growth rates (which is also consistent with the IGSCC rates given in NUREG-0313). This growth will be independent of any fluctuating loading since it is dependent only on the residual stresses from the dryer fabrication. The fuel cycle length at VYNPS, time between refueling outages, is 18-months (13,140 hrs).

The next step was to evaluate the length at which fatigue crack growth could occur. It is well established that fatigue will only occur when the threshold stress intensity range ($\Delta K_{th}$) is exceeded. For stainless steel at 288°C, this value is conservatively assumed to be 5 ksi-in$^{1/2}$. Strain gage data from an overseas BWR measured on the dryer skirt was used to determine the magnitude of the peak alternating stresses that would be present. A conservative adjustment to this peak stress for use in conjunction with the VYNPS drain channel was performed by scaling the overseas plant stresses to the steam line velocities associated with VYNPS's EPU conditions. In that the strain gage data used in this evaluation was taken from the skirt, the gage sensor is expected to represent the alternating stress distribution in the neighborhood of the observed cracking. It is also assumed that the presence of this length of cracking would not alter the structural response at the drain channel.

The results of this evaluation established that the flaw would be predicted to reach 13.3 inches after 18 months. The associated $\Delta K$ for this longer crack is below the critical $\Delta K_{th}$. Only when the crack reached 15.6 inches would the crack reach the $\Delta K_{th}$ at which fatigue crack extension could take place. This would be predicted to occur after 32 months of operation. The conclusion of the analysis is consistent with the current observations that the crack is purely IGSCC. It also supports the current disposition of the flaw. The analysis is also consistent with the field occurrences of fatigue cracking in drain channels. In those cases, the cracking initiated at the lower end of the skirt, a location where cyclic stresses could produce displacements leading to crack initiation.

Drainpipe Indication

This indication is also IGSCC. Its observed circumferential length is 3.0 inches. In that there are no significant alternating loads, the only concern is lengthening by IGSCC. Using the rate of $5x10^{-5}$ in/hr on each end, consistent with established BWRVIP growth rates over 18-months of operation, the predicted crack would reach 4.2 inches, 33% of the circumference. Secondly, even if more cracking were to occur, the pipe was inserted into the drain channel prior to welding as detailed in the drawings. This additional engagement provides another source of structural margin. Therefore, this cracking is also fully acceptable.

Vane Bank End Plate Flaws

For the case of vane bank end plate flaws, the assessment is based on the following factors: (1) it is a highly redundant structure and there is no structural consequence of the cracking and (2) postulated significant flaw extension leading to the flaw reaching the full section of the channel geometry would not create the opportunity for loose parts. The field experience supports this as-is operation decision in the context that the indication will be re-inspected at the next outage.

The dryer unit end plates, with indications, are securely attached and captured within the structure of the steam dryer bank assembly. The vertical edges of these end plates are attached to the dryer assembly with 3/16-inch fillet welds (each weld approximately 48-inches long). There were no relevant indications reported in these vertical welds. The geometric configuration of unit end plates is such that the upper and lower edges are mechanically captured by the steam dryer assembly as shown in Figure EMCB-A-1-2. The reported horizontal indications were seen in the 1.25-inch inlet side end plate flange. The vanes prevent inspection of the central end plate surface, but inspection of the outlet side end plate flanges found no indications. For the purpose of this discussion it is postulated that the end plate horizontal indications propagate across the entire 8.75-inch unit end plate width including both the inlet and outlet side flange, as shown in Figure EMCB-A-1-2. Such full width through thickness cracks would have no structural impact nor is there any concern for loose parts. The separated end plate sections, as shown in Figure EMCB-A-1-2, are still attached and will continue to function.

In summary, all of the flaws were dispositioned based on sound engineering judgment. In addition, quantitative evaluations have been performed to show that the cracking is acceptable based on IGSCC and fatigue considerations.

## Figure EMCB-A-1-1
## Schematic Representation of Drain Channel Cracking

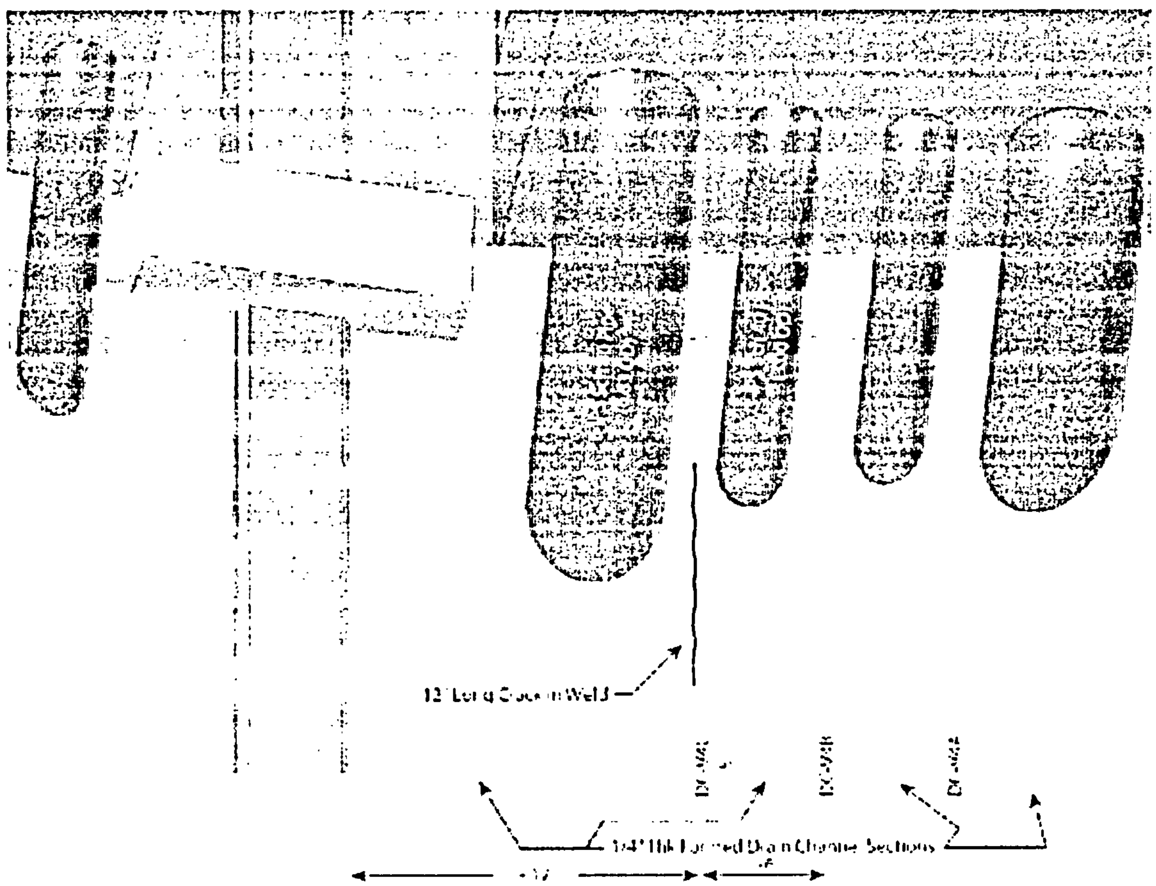

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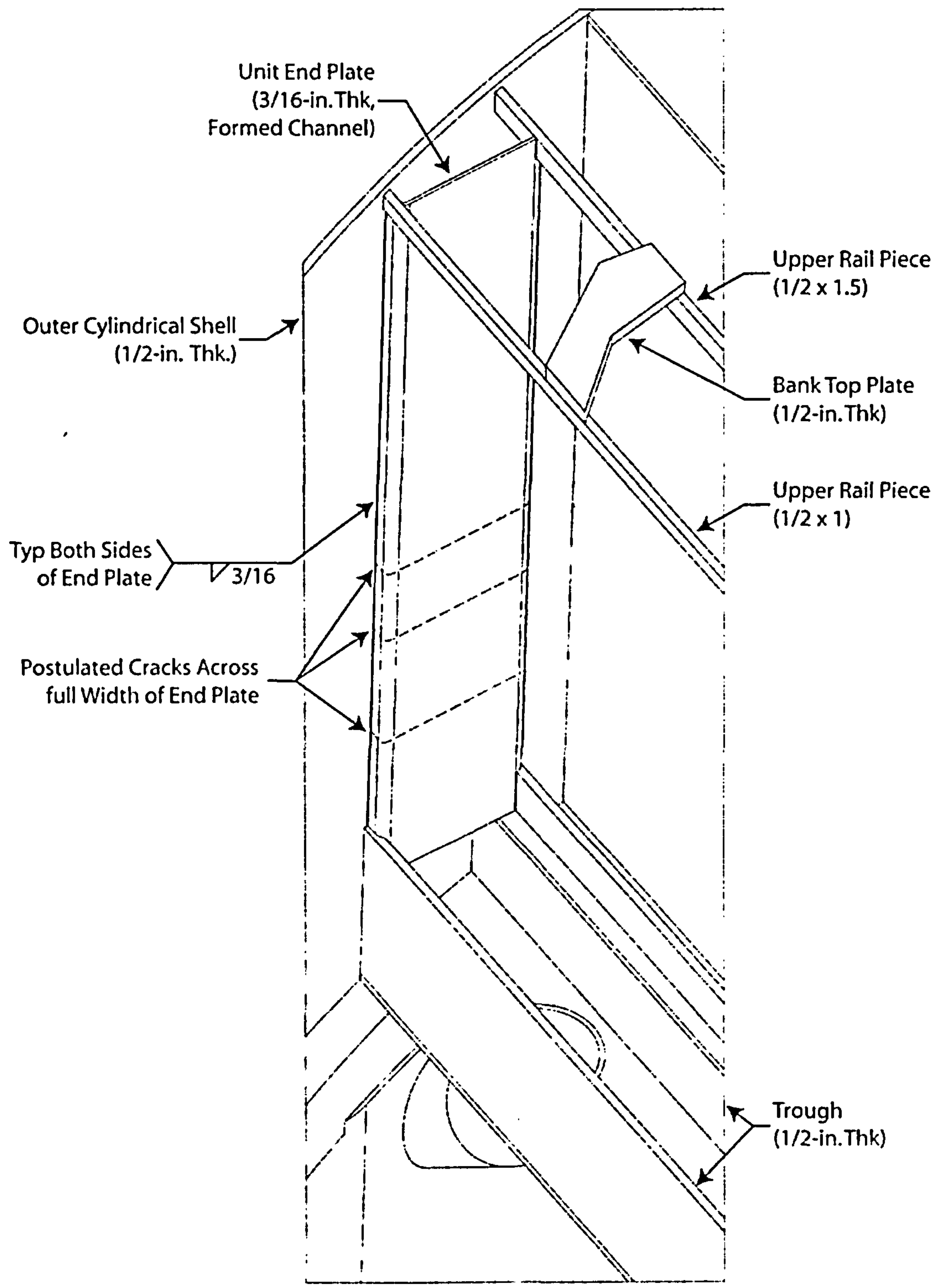
**Figure EMCB-A-1-2**
**Cut-away of Bank Showing Unit End Plate**



## RAI EMCB-A-2

During the Spring 2004 refueling outage at VYNPS, the licensee identified cracking in the steam dryer. The licensee has repaired some cracks and evaluated others acceptable for return to power at the current licensed thermal power (CLTP) level. The NRC staff requires the following information due to concerns that the proposed EPU conditions could cause the cracks to grow to a size that could affect the integrity of the steam dryer and could cause loose parts.

Provide a plan for future inspections of the steam dryer with justification relating the proposed inspection frequency to the structural integrity analysis provided in response to question 1, above.

## Response to RAI EMCB-A-2

Because the justification for leaving flaws "as-is" is provided on a cycle-by-cycle basis, the unrepaired flaws described in the response to EMCB-A RAI 1 will be inspected during each scheduled refueling outage until it is demonstrated that there is no further crack growth and the flaws have stabilized. Similarly, the repairs made to the cracked components identified during the Spring 2004 outage will be inspected during each scheduled refueling outage until the structural integrity of the repairs has been demonstrated. The dryer modifications described in the response to EMEB-B RAI 1 were designed for the life of the dryer. These modifications will be inspected during the next two scheduled refueling outages to confirm the structural integrity of the modifications. Once structural integrity of the repairs and modifications has been demonstrated and the flaws left "as-is" have been shown to have stabilized, longer inspection intervals for these locations may be justified.

The implementation of EPU will lead to an increase of the operating loads on the dryer. Therefore, the overall inspection schedule must factor this change into the plan. The current schedule for implementation of EPU at VYNPS is to perform a partial implementation, to approximately 115% of original licensed thermal power, during the current operating cycle and to implement the full EPU after the next scheduled refueling outage (fourth quarter 2005). An inspection of accessible, susceptible locations of the steam dryer will occur during the next refueling outage, presently scheduled for the fourth quarter 2005. Entergy will also inspect accessible, susceptible locations of the steam dryer during the following two scheduled refueling outages (Spring 2007 and Fall 2008). In that some of the repaired locations coincide with these high stress locations, these locations will be inspected for two cycles following full EPU implementation. Once structural integrity of the repairs and modifications has been demonstrated and the flaws left "as-is" have been shown to have stabilized at the final EPU power level, longer inspection intervals for these locations may be justified. The susceptible locations that will be inspected include any flaws left "as-is," repairs and modifications, and the high stress locations identified by the VYNPS dryer structural integrity analysis and SIL 644 Supplement 1. This inspection plan may be revised as appropriate to be consistent with any future BWRVIP Inspection and Evaluation guidelines or other industry guidance.

## Mechanical and Civil Engineering Branch (EMEB)

### RAI EMEB-B-1

Supplement 4 (Reference 5), Attachment 8, page 26, states that the planned modification of the VYNPS steam dryer includes the replacement of the upper 30-inch section of the original 0.5-inch thick flat vertical hoods (90 degree and 270 degree azimuths), with 1-inch thickness plate. It also states that gussets (33 inches high) are being added between the modified lower dryer cover plates and the unmodified section of the flat vertical dryer hoods. Confirm whether this is the current modified steam dryer installed at VYNPS. If different, describe the actual dryer modification as currently installed at VYNPS. The recent steam dryer failure at Quad Cities Unit 2 (QC2), for this type of design, with gussets attached to the unmodified section of the flat vertical dryer hoods, created stress concentration and cracks at the weld. In light of the failure of a similar modification at QC2, discuss how the steam dryer modification will ensure the structural integrity of the dryer components at VYNPS for the operation at EPU conditions.

### Response to RAI EMEB-B-1

#### 1.0 Installed Modification to VYNPS Steam Dryer

The modification physically installed to the VYNPS steam dryer during the April 2004 VYNPS refueling outage is different than that originally proposed and stated in Supplement 4 (Entergy Letter BVY 04-009 to NRC dated January 31, 2004), Attachment 8, page 26. As a result of the 2004 QC2 experience, the modifications for the VYNPS steam dryer were revised before installation to incorporate the design features that were developed for QC2 to reduce local stress concentrations. The originally proposed modification and the installed dryer modification are stated in the table below and are shown in EMEB-B-1-1 and 2. Figure EMEB-B-1-2a shows photographs of the installed modification at VYNPS.

**Table EMEB-B-1-1**
**VYNPS Dryer Modification Design Comparison**

| Dryer Component | Original Proposed Dryer Modification (Supplement 4, Attachment 8, page 26) | Installed Dryer Modification at VYNPS during April 2004 outage |
|---|---|---|
| Outer Vertical Hood Plates (90 and 270 degree) | Replace upper 30 inch section with 1-inch thick plate | Entire front vertical hood plate (61-inch high) replaced with 1-inch thick plate |
| Outer Vertical Hood Plates and Lower Horizontal Cover Plate (90 and 270 degree) | Install 3 reinforcing gussets (33-inch high) welded to Outer vertical hood plate and lower horizontal cover plate | Installed 3 reinforcing gussets (55.5-inch high) welded to Outer vertical hood plate and horizontal cover plate. |
| Lower Horizontal Cover Plate (90 and 270 degree) | Replace original ¼-inch thick plate with ½-inch thick | Replaced original ¼-inch thick plate with 5/8-inch |

| Dryer Component | Original Proposed Dryer Modification (Supplement 4, Attachment 8, page 26) | Installed Dryer Modification at VYNPS during April 2004 outage |
|---|---|---|
| | plate | thick plate |
| Upper Horizontal Cover Plate (90 and 270 degree) | Replace fifteen inch section at intersection of front vertical hoods with 1-inch thick plate | Replaced fifteen inch section at intersection of front vertical hoods with 1-inch thick plate |
| Internal Bracing Brackets at Outer Vertical Hoods | Remove Brackets | Brackets Removed |
| Dryer Bank Tie Bars | Install reinforcement brackets (1-inch by 3-inch barstock) adjacent to existing tie-bars | Replaced existing tie bars with new design. |

As discussed in Attachment 8 of Entergy letter (BVY 04-009) to NRC dated January 31, 2004, the results of the quantitative evaluation of the VYNPS steam dryer indicated that modifications to the steam dryer are necessary in order to ensure the steam dryer structural integrity under CPPU conditions. The acceptance criteria for the modified VYNPS steam dryer are as follows:

1. The maximum vibratory stress for the modified steam dryer is less than 13,600 psi (ASME Code Section III, 1986, Division 1, Appendix I, Figure I-9.2.2, Design Fatigue Curve for Austenitic Steels).
2. The primary stresses of the steam dryer components, when subjected to normal (Service Level A), upset (Service Level B) and faulted condition (Service Level D) loading, are below the following ASME Code Section III limits:

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**Table EMEB-B-1-2**
**ASME Stress Limits**

| Service level | Stress category | Stress limit |
|---|---|---|
| *Service levels A & B* | $P_m$ | $S_m$ |
| | $P_m + P_b$ | 1.5 $S_m$ |
| | Shear stress | 0.6 $S_m$ |
| | Bearing stress | 1.5 $S_y$ |
| | $\Sigma$ fatigue damage | 1.0 |
| *Service level D* | $P_m$ | min (2.4$S_m$ , 0.7$S_u$) |
| | $P_m + P_b$ | min (3.6$S_m$, 1.05$S_u$) |
| | Shear stress | 1.2 $S_m$ |
| | Bearing stress | 3.0 $S_y$ |

$P_m$: Primary membrane stress intensity

$P_b$: Primary bending stress intensity

$S_m$: Stress intensity limit

$S_y$: Yield strength

$S_u$: Ultimate strength

The rationale for the change to the installed steam dryer modification at VYNPS versus the proposed modification reported in Attachment 8, page 26 of Entergy Letter BVY 04-009 to NRC dated January 31, 2004, was to incorporate lessons learned during March 2004 from the cracking observed on the Quad Cities Unit 2 steam dryer. In March 2004, refueling outage inspections of the Quad Cities Unit 2 steam dryer indicated additional cracks in steam dryer components. Three of the six repair gussets (installed in 2003) between the lower cover plate and the outer vertical hoods had cracking at the upper tips of the gussets. These gusset cracks were determined to have been caused by the Quad Cities Unit 2 gusset installation and weld configuration where the gusset did not extend to the repaired 1-inch thick outer hood vertical plate and the gusset weld did not transition smoothly into the outer hood vertical plate.

GE Nuclear had initially performed finite element analysis of the VYNPS steam dryer that would have incorporated the proposed modification design as stated in Attachment 8, page 26 of Entergy Letter BVY 04-009 to NRC dated January 31, 2004. The results of this analysis indicated that a high stress location existed at the tips of the modification gussets, e.g., the point of intersection of the gusset tips with the dryer front vertical hood. While this high stress condition could be alleviated by the design of the replacement gussets, the results of the March 2004 steam dryer inspections at Quad Cities Unit 2, indicated that improvements to the VYNPS design should be incorporated. Specifically, a modification that included a full replacement of the outer front hood vertical plates would allow the gusset connection to the replacement front hood vertical plate by shop welding instead of underwater welding as was the case at Quad Cities Unit 2. In addition, the connection between the modification gussets and the replacement lower horizontal cover plate is performed by the installation of gusset extensions as shown in Figure EMEB-B-1-2, detail C.

Even though VYNPS inspection results have shown no cracking of dryer bank tie-bars, the VYNPS steam dryer modification includes replacing the original dryer bank tie bars with a modified design. The modified tie bars for VYNPS are similar in design to the horizontal tie bars installed in March 2004 at Quad Cities Unit 2. The modified tie bar design allows the tie-bar horizontal pads to be welded to the top hood plates in adjacent dryer banks. The dryer loading conditions that have led to previous tie-bar cracking at boiling water reactors are not fully understood. The expected maximum stresses at the tie-bar attachment weld locations are predicted to be approximately 3 times lower than previous tie bar designs. The fundamental frequencies of the modified tie-bar designs are sufficiently removed from the expected acoustic excitation frequencies that resonant excitation of the modified tie bar is not expected to be a problem at CPPU operating conditions.

To ensure that the modified steam dryer at VYNPS is acceptable for operation at CPPU conditions, quantitative evaluations using ANSYS finite element analyses (FEA) models of the modified VYNPS steam dryer were performed. The FEA model for the unmodified VYNPS steam dryer is shown in Figure EMEB-B-1-14. The FEA model for the modified VYNPS steam dryer is shown in Figure EMEB-B-1-15 and 16. These evaluations fall into two categories: (1) Fatigue evaluations for acceptability against flow induced vibration and (2) Structural integrity evaluations against primary stress levels for service level A, B, and D conditions.

## 2.0 Fatigue Evaluation of Modified Steam Dryer

VYNPS Modified Steam Dryer Flow Induced Vibration Evaluation

The VYNPS dryer proactive strengthening design modification is based on the "Equivalent Static" method for EPU conditions. A revision to this analysis of record utilized a more refined dynamic response methodology. The "Dynamic Response Spectrum" method refines the structural response to more accurately represent the dynamic effects of flow induced vibration (FIV). The results of the Dynamic Response Spectrum analysis do not show any additional modifications are required in order for the VYNPS steam dryer to operate at EPU conditions.

Dynamic Response Spectrum Method

A dynamic response spectrum method was performed for the steam dryer FIV-induced fatigue susceptibility evaluation. This method consists of the following major steps:

(a) Start with pressure loads specified as a function of frequency based on the measured data from plants (reference plant pressure spectrum as shown in Figure 2 of Reference 1). The magnitudes of the reference plant pressure spectrum are adjusted based on the ratio of the VYNPS steam line velocity to the reference plant streamline velocity for the CLTP and CPPU conditions, respectively. This adjusting process is the same as described in Section 4.1.1 of Reference 1, with the pressure spectrum for each reference plant measured data adjusted separately to the VYNPS conditions. The scaling results using the Reference load amplitudes to derive the VYNPS load amplitudes for both CLTP and CPPU conditions are shown in Figures EMEB-B-1-3 and 4.

(b) Generate a synthetic pressure time history based on the adjusted VYNPS pressure spectrum using the [[ ]]. Next, perform a [[ ]] of the newly generated discrete pressure time history, which transforms the pressure from the time domain back to frequency domain. This pressure spectrum is compared with the input adjusted pressure spectrum as defined in Step (a) to ensure the accurate representation in either time or frequency domain of the given pressure loading.

(c) Calculate a response spectrum based on the synthetic pressure time history using computer program SPECA05V. Broadening of the response spectrum was made to account for the uncertainty in frequency. A broadening of [[ ]] is selected in the present application. Also a damping ratio of [[ ]] is used for the steam dryer evaluation.

(d) Repeat Step (a) through Step (c) for the measured pressure load of a different plant and/or a different pressure gage.

(e) Enveloping and further broadening all the response spectra as generated from the previous steps is then performed. This final response spectrum is the input to the ANSYS computer program to perform the steam dryer structural evaluation.

Shown in Figures EMEB-B-1-5 through 7 are pressure spectra, synthetic pressure time histories, and response spectra from the peak pressure measurements of three plants (one domestic plant and two foreign plants) scaled to the VYNPS CLTP condition. Figure EMEB-B-1-8 presents the enveloping and further broadened response spectrum.

Similarly, shown in Figures EMEB-B-1-9 through 12 are three response spectra, synthetic pressure time histories, response spectra and the enveloping response spectrum for the VYNPS CPPU condition. Note that, in the figures, the input pressure spectra are given in rms amplitude. The corresponding synthetic pressure time histories, response spectra and enveloping response spectrum are expressed in Peak amplitude, where a peak factor of [[ ]] to convert rms amplitude to Peak amplitude has been incorporated.

Based on the response spectrum method, ANSYS calculated component stress distributions for the VYNPS unmodified and modified steam dryer at CLTP and CPPU conditions are calculated. The SRSS (square root of the sum of the squares) method of the ANSYS modal combination is used to obtain these stress results.

Next, the peak stresses calculated by ANSYS for each steam dryer component are multiplied by the appropriate stress concentration factors to account for undersized welds and weld quality factor ([[ ]]) to arrive at the peak dynamic stress at the corresponding locations. The calculation of the stress concentration factor to account for undersized welds is the [[ ]]. For example, in the modified dryer stress analysis, a 1/2--inch fillet weld is used for the 5/8- inch thick lower cover plate; the stress factor to convert the plate stress to the fillet weld stress is [[ ]]. The other components with undersized welds in the modified steam dryer are the outer hood top plate [[ ]], outer hood vertical plate bottom and top welds [[ ]], and gusset extension weld to lower cover plate [[ ]]. Full penetration welds are used in the VYNPS steam dryer only for the inner hood top plate, inner hood vertical plates and the inner hood brackets. All other welds are either fillet or butt welds.

Under CLTP condition, of all the outer bank components, the highest stress is at the outer cover plate weld. By scaling this outer cover plate weld stress to the fatigue failure criterion of 27,200 psi, a scaling factor of 0.03446 (1÷29) is calculated. This 0.03446 scaling factor is then applied to calculate the dynamic stresses of all other components, and also to calculate the dynamic stresses under CPPU condition. The use of the scaling factor is considered appropriate since inspections of the VYNPS steam dryer in April 2004 did not indicate any cracking of the outer cover plate. The uncorrected stress levels calculated for the unmodified VYNPS steam dryer at CLTP operating conditions would have resulted in extensive degradation of the steam dryer after a very short time period following the original licensing of the plant. In actuality, the steam dryer has been operating for greater than thirty years without any cracking of the outer cover plate.

Table EMEB-B-1-3 lists the peak dynamic stresses of the VYNPS stream dryer under CLTP and CPPU conditions. The results of the dynamic response spectrum analysis confirm that the modifications to the VYNPS steam dryer are adequate for CPPU steam conditions and that there are no additional steam dryer component vulnerabilities from those determined from the equivalent static analysis method presented in Reference 1 for component vulnerability screening.

NON-PROPRIETARY INFORMATION

## Table EMEB-B-1-3
## VYNPS Steam Dryer Response Spectrum Analysis

| Component | Response Spectrum Analysis | | |
|---|---|---|---|
| | **Unmodified Dryer CLTP Dynamic Loading (psi)** | **Modified Dryer CLTP Dynamic Loading (psi** | **Modified Dryer EPU Dynamic Loading (psi)** |
| **Horizontal plates:** | | | |
| Base plate | [[ | | |
| Outer cover plate | | | |
| Outer Hood top plates | | | |
| **Vertical plates:** | | | |
| Outer Hood vertical plates | | | |
| Inner Hood end plates | | | |
| Outer Hood end plates | | | |
| Outer Hood Brackets (gussets) Removed by Modification | | | |
| Hood below cover plate | | | |
| Steam 'dam' | | | |
| Steam 'dam' gussets | | | |
| **Other Plates** | | | |
| Hood partition plates | | | |
| Baffle plates | | | |
| Outlet plenum ends | | | |
| **Ring, Beams & Gussets** | | | |
| Dryer support ring | | | |
| Bottom cross beams | | | |
| Cross beam gussets | | | |
| **Gussets Modification** | | | |
| Gusset to Cover plate and Front Hood | | | |
| Gusset Extension Weld to lower cover plate | | | ]] |

Note 1: For the modified steam dryer, the outer cover plate is installed perpendicular to the replaced outer hood vertical plate. Therefore there is no hood below cover plate component in the modified steam dryer.

Equivalent Static Method Evaluation of Modified VYNPS Steam Dryer

The steam dryer load definition process and derivation, as well as the VYNPS steam dryer fluctuating load input calculation was previously supplied to the NRC in Sections 4 and Sections 6 of Reference 1. Table EMEB-B-1-4 provides the VYNPS fluctuating load input.

**Table EMEB-B-1-4**
**VYNPS Dryer FIV Load Input**

| Frequency Range (Hz) | VYNPS CLTP FIV Load Amplitude rms psi | VYNPS CPPU FIV Load Amplitude rms psi |
|---|---|---|
| 0 to 55 | [[ | |
| 55 to 120 | | |
| 120 to 205 | | |
| 205 to 320 | | |
| 320 to 525 | | |
| 525 to 800 | | ]] |

In order to provide a comparison of the analysis presented in Reference 1 for the VYNPS unmodified steam dryer to the modified steam dryer, GENE applied the same process used in Reference 1, Section 1 to evaluate the steam dryer dynamic vibration response to assess the vulnerability to FIV-induced fatigue. This process is based on the VYNPS specific scaled load definition shown above, the natural frequency assessment based on the ANSYS dryer model of the modified VYNPS steam dryer and the resultant stresses based on the application of a normalized pressure load to all pressure bearing surfaces. The method is termed "Equivalent Static Analysis Method." The Equivalent Static Analysis Method for the VYNPS modified dryer evaluation consists of the following process steps:

1. A Finite Element Analysis (FEA) model of the modified VYNPS steam dryer was developed. This model was constructed using VYNPS specific dryer dimensions and material properties.
2. The FEA computes steam dryer component natural frequencies and mode shapes.
3. A unit static pressure load is applied in the FEA model. Steam Dryer component Membrane (Pm) and Surface (Pm + Pb) stresses are computed from the applied unit load.
4. Dynamic Loading (DL) on the steam dryer components is computed via the following equation:

   DL = (Pm+Pb) x (FIV Load rms) x (P) x (AF) x (C) x (WUF) x (DPR)

   Where:

   DL = Dynamic Stress (psi)

   Pm+Pb = Surface stress computed from [[ ]] static load in the FEA model

FIV Load rms = Fluctuating load (Root-mean-squared (rms) load amplitude) obtained from plant measured data and scaled to VYNPS steam velocity for CLTP and CPPU conditions.

P = Peak factor for load to convert rms amplitude to Peak amplitude. For a pure single frequency sinusoidal time function, the peak is equal to √(2) times the rms amplitude. For the flow induced vibration time function of reactor internal components, a factor of [[ ]] is used to account for the summation of many frequencies.

AF = Amplification Factor or Dynamic Load factor. Factor can vary from [[ ]] depending on the degree of matching between a natural frequency and a spectral peak. For the evaluation of the modified steam dryer a factor of [[ ]] is used to conservatively assume a high degree of matching.

C = Stress Concentration Factor including the weld quality factor. The FEM calculated peak stress has picked up some of the stress concentration factor. A C value of [[ ]] is used for butt and fillet welds based on good shop quality welds and the inspection techniques typically used in dryer fabrication. A C value of [[ ]] is used for full penetration welds. This is based on Table NG-3352-1 of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, 1989 Edition with no Addenda, where the fatigue factor for a penetration weld is 1/2 of fillet weld factor using visual inspection.

WUF = Weld Undersize Factor. This is defined as the square of the plate thickness to the weld size ratio. [[ ]]

DPR = Dynamic Pressure Ratio. Based on Computation Fluid Dynamic (CFD) modeling of the VYNPS steam dryer, the static pressure loading on the steam dryer inner banks is significantly lower (approximately [[ ]]) than the static pressure loading on the steam dryer inner hood components. A minimum conservative DPR, minimum of [[ ]], is applied to the inner bank components

Table EMEB-B-1-5 shows the modified and unmodified VYNPS steam dryer alternating stresses calculated by the equivalent static method. Large reductions in the alternating stresses (greater than factor of 10) are seen for the modified steam dryer front hood and cover plate. This is due to a combination of reduced maximum surface fiber stress because of the thicker plate material and removal of the high stress concentration from the hood brackets as well as a shift of the component natural frequency.

NON-PROPRIETARY INFORMATION

## Table EMEB-B-1-5
## Comparison of Modified and Unmodified VYNPS Steam Dryer Alternating Stresses – Equivalent Static Method

| Component | Unmodified dryer Maximum Surface Fiber Stress (psi) | Modified dryer Maximum Surface Fiber Stress (psi) | Unmodified dryer Associated Frequency (Hz) | Modified dryer Associated Frequency (Hz) | Unmodified dryer CLTP Maximum Stress (psi) | Modified dryer CLTP Maximum Stress (psi | Unmodified dryer CPPU Maximum Stress (psi) | Modified dryer CPPU Maximum Stress (psi) | Dynamic Pressure Ratio for Modified Dryer Evaluation |
|---|---|---|---|---|---|---|---|---|---|
| **Horizontal plates:** | | | | | | | | | |
| Base plate | [[ | | | | | | | | |
| Outer cover plate | | | | | | | | | |
| Outer cover plate | | | | | | | | | |
| Outer Hood top plates | | | | | | | | | |
| **Vertical plates:** | | | | | | | | | |
| Outer Hood vertical plates | | | | | | | | | |
| Inner Hood end plates | | | | | | | | | |
| Outer Hood end plates | | | | | | | | | |
| Outer Hood Brackets (gussets) Removed by Modification | | | | | | | | | |
| Hood below cover plate | | | | | | | | | |
| Steam 'dam' | | | | | | | | | |
| Steam 'dam' gussets | | | | | | | | | |
| **Other Plates** | | | | | | | | | |
| Hood partition plates | | | | | | | | | |
| Baffle plates | | | | | | | | | |
| Outlet plenum ends | | | | | | | | | |

NON-PROPRIETARY INFORMATION

## Table EMEB-B-1-5 (cont)
## Comparison of Modified and Unmodified VYNPS Steam Dryer Alternating Stresses – Equivalent Static Method

| Component | Unmodified dryer Maximum Surface Fiber Stress (psi) | Modified dryer Maximum Surface Fiber Stress (psi) | Unmodified dryer Associated Frequency (Hz) | Modified dryer Associated Frequency (Hz) | Unmodified dryer CLTP Maximum Stress (psi) | Modified dryer CLTP Maximum Stress (psi | Unmodified dryer CPPU Maximum Stress (psi) | Modified dryer CPPU Maximum Stress (psi) | Dynamic Pressure Ratio for Modified Dryer Evaluation |
|---|---|---|---|---|---|---|---|---|---|
| **Ring, Beams & Gussets** | | | | | | | | | |
| Dryer support ring | | | | | | | | | |
| Bottom cross beams | | | | | | | | | |
| Cross beam gussets | | | | | | | | | |
| **Gussets Modification** | | | | | | | | | |
| Gusset to Cover plate and Front Hood | | | | | | | | | |
| Gusset Extension Weld to lower cover plate | | | | | | | | | ]] |

Note 1: For the modified steam dryer, the outer cover plate is installed perpendicular to the replaced outer hood vertical plate. Therefore there is no hood below cover plate component in the modified steam dryer.

Note 2: During the analysis for the modified steam dryer, it was noted that maximum surface fiber stress [[ ]] reported in Reference 1 for the unmodified steam dryer was in error. The actual maximum surface fiber stress for this component is [[ ]], both for the unmodified and modified steam dryer. With this corrected surface fiber stress, the partition plates are not a critical dryer component. The unmodified dryer maximum stress at CPPU conditions is [[ ]].

## 3.0 Structural Integrity Evaluations Against Primary Stress Levels for Service Level A, B, And D Conditions.

The VYNPS steam dryer modification design was evaluated using the ASME Code, Reference 2, as a design guide although the dryer is not an ASME Code component. Specifically, structural adequacy for Service Level A and B loads was investigated using the corresponding stress limits of Reference 2 with the exception of application of the weld quality factors. Weld quality factors are described in the ASME code Table NG-3352-1 for safety components, such as the reactor pressure vessel, that contain radioactive fluid. Because the steam dryer is not a safety-related pressure-retaining component [[ ]]. The requirement of 'no loose parts' during Service Level D events was investigated using stress limits of Subsection NG and Appendix F of the ASME Code, Reference 3. (Note that for completeness, application of the seismic loading in the [[ ]] was considered).

Load Combinations for Stress Evaluation

The VYNPS steam dryer was originally procured and supplied as a non-safety related, non-seismic category I, non-ASME component. For the original design of the VYNPS Steam Dryer, the following service condition and acceptance criterion were stated:

- The principal design loads considered in the analysis of the steam dryer assembly are the weight loads and the pressure loads, which are present during accident conditions.

- In the event of a guillotine steam line break outside the drywell, dryer design must preclude the possibility of dryer debris entering the steam line and interfering with isolation valve closure.

- The structural elements, which hold the dryer in place, are designed to accommodate the pressure loading due to a break outside the isolation valves within the ASME Code, Section III stress criteria. The flat panels, which form partitions in the dryer, are designed so that the elastic collapse loading on these panels is not exceeded under these same pressure loadings.

The above criteria continue as the design basis for the VYNPS steam dryer, both at CLTP and CPPU conditions. The VYNPS steam dryer design basis continues to be maintaining structural integrity after a steam line break outside of containment.

However, it was considered prudent to perform additional structural evaluations on the modified steam dryer in order to confirm the robustness of the modification. This is due to the increased scrutiny on steam dryer structural integrity in light of recent dryer experience at CPPU conditions. The major consideration is the inclusion of Upset case loading combinations for the modified steam dryer. The steam dryer evaluation process has indicated that the structural loading on the dryer during normal and upset conditions, either at CLTP or CPPU, is more complex than had previously been considered. The concern is that Upset events are moderate frequency events from which a given plant should be able to recover, e.g. expeditiously restart the unit after the event has occurred. From a review of past evaluations on steam dryers, GE has realized that there are different loading combinations on the steam dryer that are both realistic and that could question the validity of the assumption that the plant can restart following an Upset event without performing inspection of the steam dryer. Table EMEB-B-1-6 provides the load combinations and load cases used in the VYNPS dryer analysis.

**Table EMEB-B-1-6**
**ASME Code Section III Load Combinations**

| Case | Service Level | Load Combination |
|---|---|---|
| [[ | | ]] |

OBE: Operational Basis Earthquake

TSV: Turbine Stop Valve Fast Closure Loading

SSE: Safe Shutdown Earthquake

Dryer Loads for Steam Dryer Structural Evaluations

Seismic Loads

The following seismic loads were used in the structural evaluation. These seismic loads are unchanged with CPPU and were used in the structural evaluation of RPV internals for CPPU. The accelerations are listed below.

| | | |
|---|---|---|
| OBEX=0.4g | OBEY=0.4g | OBE vertical=0.22g |
| SSEX=0.63g | SSEY=0.63g | SSE vertical=0.31g |

Because the modified dryer first mode frequency is in the ZPA (Zero Period Acceleration) range of the seismic load (higher than 60 Hz), the time history maximum acceleration (g) load is the dryer acceleration (g) load for equivalent static analysis. (Note that for completeness, application of the seismic loading in the [[ ]] was considered).

Pressure Loads

The pressure differentials across the steam dryer are calculated for three categories of events; normal, upset, and faulted conditions. Normal conditions are the steady-state operating conditions. Upset conditions are the anticipated transient events. The upset category is further divided into two sub-categories: forward flow (e.g. one stuck open relief valve) and backward flow (e.g. turbine stop valve closure). Faulted conditions are the design basis accident events (e.g. main steam line break). Upon occurrence of Turbine Stop Valve (TSV) closure transient, a pressure wave is created that travels at sonic velocity through each Main Steam (MS) line to the Reactor Pressure Vessel (RPV) and the steam dryer. Repeated reflection of the pressure wave at the RPV and TSV produces pressure time varying function on the steam dryer front hoods. Typical pressure distribution and time histories on the front hoods due to TSV closure are shown in Figure EMEB-B-1-13. Because the wave load due to TSV closure is an impulse load, a dynamic load factor of 1.5[1] is applied based on the impulse shape and the natural frequency of the front hood.

The pressure differentials across the steam dryer for the normal conditions at CPPU power level are summarized in Table EMEB-B-1-7.

**Table EMEB-B-1-7**
**Steam Dryer Pressure Differentials for Normal Conditions at CPPU**

| Description | Value(Note 1) (psid) |
|---|---|
| Vertical Cover Plate | 2.45 |
| Horizontal Cover Plate | 1.91 |
| Horizontal Section of Outer Hood | 0.57 |
| Horizontal Section of Inner Hood | 0.43 |
| Vertical Section of Inner Hood | 0.11 |

Note: 1) The results are from Computation Fluid Dynamics (CFD) modeling of the VYNPS Dryer operating at CPPU conditions.

The pressure differentials across the steam dryer due to forward flow for upset conditions at CPPU power level are summarized in Table EMEB-B-1-8.

[1] Figure 2.8 of Biggs, John M., Introduction to Structural Dynamics, New York: McGraw Hill, 1964

NON-PROPRIETARY INFORMATION

**Table EMEB-B-1-8**
**Steam Dryer Pressure Differential for Forward Flow Upset Conditions at CPPU**

| Description | Value (psid) |
|---|---|
| Vertical Cover Plate | 5.07 |
| Horizontal Cover Plate | 3.95 |
| Horizontal Section of Outer Hood | 1.18 |
| Horizontal Section of Inner Hood | 0.89 |
| Vertical Section of Inner Hood | 0.23 |

The maximum acoustic loads, e.g. backward flow, on the dryer face at CPPU power level are summarized in Table EMEB-B-1-9. Typical pressure time history is shown in Figure EMEB-B-1-11.

**Table EMEB-B-1-9**
**Maximum Acoustic Load on the Dryer Face at CPPU**

| *y, Dryer Vertical Centerline* | Pressure Differential *(psid)* | | | | | |
|---|---|---|---|---|---|---|
| *5.16 ft* | 1.20 | 0.66 | 0.70 | 0.70 | 0.66 | |
| *4.128 ft* | 2.47 | 2.45 | 2.41 | 2.29 | 1.75 | |
| *3.096 ft* | 4.95 | 4.54 | 4.47 | 4.61 | 3.84 | |
| *2.064 ft* | 6.36 | 6.45 | 6.29 | 5.97 | 4.86 | |
| *1.032 ft* | 7.60 | 7.79 | 7.81 | 7.70 | 6.58 | |
| *0.0 ft* | 8.33 | 8.24 | 8.60 | 8.19 | 7.11 | |
| *Coordinate (x,y)* | *0.0 ft* | *1.075 ft* | *2.15 ft* | *3.225 ft* | *4.3 ft* | *x, Lower Horizontal Cover Plate* |

The upset flow-induced load on the dryer face at CPPU is 15.7 psid. This is the maximum load on the dryer face that is directly opposite from the steam line nozzle (the projected area of the steam line nozzle on to the dryer face).

In addition, the pressure differential across the steam dryer vanes due to forward flow for faulted conditions at CPPU for a steam line break outside containment is 6.9 psi.

Evaluation of Primary Stresses under ASME Code Section III Loads

Each load combination tabulated in Table EMEB-B-1-6 was analyzed for the modified dryer as well as the unmodified dryer. In the ASME Code Section III load analysis, the dynamic loads, such OBE, SSE and TSV loads should be combined by square root of the sum of the square (SRSS). This analysis combined the dynamic loads by algebraic sum. Because the OBE and SSE have been input in both the positive and negative directions and both results are compared with the allowable limits, the results are equivalent to absolute sum results. Therefore, the load combinations are conservative. The TSV pressure impulse load has been multiplied by a dynamic load factor of 1.5 in the input for ANSYS analyses.

Table EMEB-B-1-10 summarizes the stresses for the unmodified dryer. All the stresses meet the allowable limits.

The main focus of this analysis is the modified dryer. Because there are undersized welds in higher stress locations, Tables EMEB-B-1-11 through 17 tabulate the stresses for each weld with the undersize weld factor included. The maximum stress ratio is 0.676, at the gusset, due to Service Level B-3 load combination. There is more than 30% of margin for ASME load combination.

The results of the evaluation show that each weld for the modified dryer meets the ASME primary stress allowable limit. Since the TSV loading has about 400 cycles, a check of the fatigue usage was performed and found the cumulative usage factors are less than 0.05 for all the welds.

## Table EMEB-B-1-10
## Maximum Primary Stresses for Unmodified Dryer

| Unmodified Dryer<br>Operating Condition<br>Finite element stress type | Stress Intensity, psi (Note 2)<br>Hood Vertical Plate (Note 3)<br>Local membrane stress | Hood Vertical Plate (Note 3)<br>Local surface maximum stress | Lower Cover Plate weld (Note 1)<br>Local membrane stress | Lower Cover Plate weld (Note 1)<br>Local surface maximum stress |
|---|---|---|---|---|
| ASME classification (Note 2) | $P_m+P_b$ | $P_m+P_b$ + Secondary Q | $P_m+P_b$ | $P_m+P_b$ + Secondary Q |
| [[ ]] Pressure case | 1044 | 11059 | 3040 | 20974 |
| Service level A | 2209 | 25793 | 7595 | 51239 |
| Service level B (note 4) | 3617 | 52270 | 20174 | 126,565 |
| Service level D | 4260 | 71386 | 21038 | 146,775 |

Notes

1. The maximum stress occurs at the cover plate weld to the front hood. The under size factor equals [[ ]]. For bending stress the stress times the factor of [[ ]].
2. The maximum stress occurs at the weld between the internal braces to the front hood. This is a localized bending stress, which is classified as a secondary stress in accordance with ASME Code Table NB-3217-1 item for nozzle wall bending.
3. The stresses are selected from outer hood stresses, including 1-psi case.
4. Dynamic load factor of 1.5 is applied for turbine stop valve closure pressure loads.

---

1) $P_m$: membrane stress; $P_b$: bending stress
2) Stress Limits

| | | |
|---|---|---|
| Service Levels A/B | $S_m = 13995$ | $1.5S_m = 20992$ |
| Service Level D | $2.4S_m = 33588$ | $3.6S_m = 50381$ |

3) Listed stresses are the maximum stresses anywhere in the components rather than section-averaged stresses. All stresses listed are plate stresses. The membrane stress, $P_m$, listed in this table is the maximum membrane stress and not the general membrane stress. $P_m + P_b$ stresses shown are also maximum. The above limits apply to general membrane stress and general membrane plus bending stress. These maximum stresses are generally considered in fatigue evaluations.

NON-PROPRIETARY INFORMATION

## Table EMEB-B-1-11
## Modified Dryer Outer Cover Plate ASME Primary and Secondary Stresses

| Item | Service Level | Load Case | (A) Local membrane stress (psi) | (B) Surface maximum stress (psi) | Plate thickness (inch) | Fillet weld size (inch) | (C) Undersized Weld stress factor | $P_m+P_b$ (A) x (C) at weld (psi) | Local membrane Allowable stress (psi) | (D) Primary stress ratio | $P_m + P_b + Q$ stress, (B)x(C) (psi) | Alternating stress, Salt (psi) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | [[ ]] psi | | 366 | 2294 | 0.625 | 0.500 | 1.56 | 572 | 20588 | N/A | 2294 | N/A |
| 2 | Level A | 1 | 818 | 5660 | 0.625 | 0.500 | 1.56 | 1278 | 20588 | 0.062 | 5660 | 10188 |
| 3 | Level B | 1 | 1721 | 11414 | 0.625 | 0.500 | 1.56 | 2689 | 20588 | 0.131 | 11414 | 20545 |
| 4 | Level B | 2 | 1842 | 12114 | 0.625 | 0.500 | 1.56 | 2878 | 20588 | 0.140 | 12114 | 21805 |
| 5 | Level B | 3 | 3072 | 14609 | 0.625 | 0.500 | 1.56 | 4800 | 20588 | 0.233 | 14609 | 26296 |
| 6 | Level B | 4 | 2988 | 14089 | 0.625 | 0.500 | 1.56 | 4669 | 20588 | 0.227 | 14089 | 25360 |
| 7 | Level B | 5 | 1216 | 6059 | 0.625 | 0.500 | 1.56 | 1900 | 20588 | 0.092 | 6059 | 10906 |
| 8 | Level B | 6 | 878 | 6010 | 0.625 | 0.500 | 1.56 | 1372 | 20588 | 0.067 | 6010 | 10818 |
| 9 | Level D | 1 | 2841 | 15176 | 0.625 | 0.500 | 1.56 | 4439 | 49410 | 0.090 | 15176 | 27317 |
| 10 | Level D | 2 | 3036 | 15544 | 0.625 | 0.500 | 1.56 | 4744 | 49410 | 0.096 | 15544 | 27979 |

NON-PROPRIETARY INFORMATION

## Table EMEB-B-1-12
## Original Front Hood Strip ASME Primary and Secondary Stresses

| Item | Service Level | Load Case | (A) Local membrane stress (psi) | (B) Surface maximum stress (psi) | Plate thickness (inch) | Fillet weld size (inch) | (C) Undersized Weld stress factor | $P_m+P_b$ (A) x (C) at weld (psi) | Local membrane Allowable stress (psi) | (D) Primary stress ratio | $P_m + P_b + Q$ stress, (B)x(C) (psi) | Alternating stress, Salt (psi) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | [[ ]] psi | | 820 | 1296 | 0.500 | 0.500 | 1.00 | 820 | 20588 | N/A | 1296 | N/A |
| 2 | Level A | 1 | 1831 | 3075 | 0.500 | 0.500 | 1.00 | 1831 | 20588 | 0.089 | 3075 | 5535 |
| 3 | Level B | 1 | 3776 | 6372 | 0.500 | 0.500 | 1.00 | 3776 | 20588 | 0.183 | 6372 | 11470 |
| 4 | Level B | 2 | 4069 | 6705 | 0.500 | 0.500 | 1.00 | 4069 | 20588 | 0.198 | 6705 | 12069 |
| 5 | Level B | 3 | 4847 | 5067 | 0.500 | 0.500 | 1.00 | 4847 | 20588 | 0.235 | 5067 | 9121 |
| 6 | Level B | 4 | 4544 | 4780 | 0.500 | 0.500 | 1.00 | 4544 | 20588 | 0.221 | 4780 | 8604 |
| 7 | Level B | 5 | 1700 | 2345 | 0.500 | 0.500 | 1.00 | 1700 | 20588 | 0.083 | 2345 | 4221 |
| 8 | Level B | 6 | 1973 | 3243 | 0.500 | 0.500 | 1.00 | 1973 | 20588 | 0.096 | 3243 | 5837 |
| 9 | Level D | 1 | 5111 | 6948 | 0.500 | 0.500 | 1.00 | 5111 | 49410 | 0.103 | 6948 | 12506 |
| 10 | Level D | 2 | 5588 | 7219 | 0.500 | 0.500 | 1.00 | 5588 | 49410 | 0.113 | 7219 | 12994 |

NON-PROPRIETARY INFORMATION

## Table EMEB-B-1-13
## Modified Front Hood ASME Primary and Secondary Stresses

| Item | Service Level | Load Case | (A) Local membrane stress (psi) | (B) Surface maximum stress (psi) | Plate thickness (inch) | Fillet weld size (inch) | (C) Undersized Weld stress factor | $P_m+P_b$ (A) x (C) at weld (psi) | Local membrane Allowable stress (psi) | (D) Primary stress ratio | $P_m + P_b + Q$ stress, (B)x(C) (psi) | Alternating stress, Salt (psi) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | [[ ]] psi | | 365 | 988 | 1.000 | 0.500 | 4.00 | 1460 | 20588 | N/A | 3952 | N/A |
| 2 | Level A | 1 | 903 | 2473 | 1.000 | 0.500 | 4.00 | 3612 | 20588 | 0.175 | 9892 | 17806 |
| 3 | Level B | 1 | 1082 | 4966 | 1.000 | 0.500 | 4.00 | 4328 | 20588 | 0.210 | 19864 | 35755 |
| 4 | Level B | 2 | 1925 | 5262 | 1.000 | 0.500 | 4.00 | 7700 | 20588 | 0.374 | 21048 | 37886 |
| 5 | Level B | 3 | 2787 | 5243 | 1.000 | 0.500 | 4.00 | 11148 | 20588 | 0.541 | 20972 | 37750 |
| 6 | Level B | 4 | 2686 | 5048 | 1.000 | 0.500 | 4.00 | 10744 | 20588 | 0.522 | 20192 | 36346 |
| 7 | Level B | 5 | 1213 | 2935 | 1.000 | 0.500 | 4.00 | 4852 | 20588 | 0.236 | 11740 | 21132 |
| 8 | Level B | 6 | 964 | 2621 | 1.000 | 0.500 | 4.00 | 3856 | 20588 | 0.187 | 10484 | 18871 |
| 9 | Level D | 1 | 2586 | 6341 | 1.000 | 0.500 | 4.00 | 10344 | 49410 | 0.209 | 25364 | 45655 |
| 10 | Level D | 2 | 2783 | 6660 | 1.000 | 0.500 | 4.00 | 11132 | 49410 | 0.225 | 26640 | 47952 |

NON-PROPRIETARY INFORMATION

## Table EMEB-B-1-14
## Unmodified Outer Top Hood ASME Primary and Secondary Stresses

| Item | Service Level | Load Case | (A) Local membrane stress (psi) | (B) Surface maximum stress (psi) | Plate thickness (inch) | Fillet weld size (inch) | (C) Undersized Weld stress factor | $P_m+P_b$ (A) x (C) at weld (psi) | Local membrane Allowable stress (psi) | (D) Primary stress ratio | $P_m + P_b + Q$ stress, (B)x(C) (psi) | Alternating stress, Salt (psi) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | [[ ]] psi | | 239 | 671 | 0.500 | 0.500 | 1.00 | 239 | 20588 | N/A | 671 | N/A |
| 2 | Level A | 1 | 463 | 1515 | 0.500 | 0.500 | 1.00 | 463 | 20588 | 0.022 | 1515 | 2727 |
| 3 | Level B | 1 | 1041 | 2873 | 0.500 | 0.500 | 1.00 | 1041 | 20588 | 0.051 | 2873 | 5171 |
| 4 | Level B | 2 | 1060 | 2981 | 0.500 | 0.500 | 1.00 | 1060 | 20588 | 0.051 | 2981 | 5366 |
| 5 | Level B | 3 | 1006 | 5051 | 0.500 | 0.500 | 1.00 | 1006 | 20588 | 0.049 | 5051 | 9092 |
| 6 | Level B | 4 | 1013 | 4943 | 0.500 | 0.500 | 1.00 | 1013 | 20588 | 0.049 | 4943 | 8897 |
| 7 | Level B | 5 | 640 | 1512 | 0.500 | 0.500 | 1.00 | 1013 | 20588 | 0.049 | 1512 | 2722 |
| 8 | Level B | 6 | 495 | 1569 | 0.500 | 0.500 | 1.00 | 640 | 20588 | 0.031 | 1569 | 2824 |
| 9 | Level D | 1 | 1591 | 4517 | 0.500 | 0.500 | 1.00 | 495 | 49410 | 0.010 | 4517 | 8131 |
| 10 | Level D | 2 | 1624 | 4603 | 0.500 | 0.500 | 1.00 | 1591 | 49410 | 0.032 | 4603 | 8285 |

NON-PROPRIETARY INFORMATION

## Table EMEB-B-1-15
## Modified Top Hood ASME Primary and Secondary Stresses

| Item | Service Level | Load Case | (A) Local membrane stress (psi) | (B) Surface maximum stress (psi) | Plate thickness (inch) | Fillet weld size (inch) | (C) Undersized Weld stress factor | $P_m+P_b$ (A) x (C) at weld (psi) | Local membrane Allowable stress (psi) | (D) Primary stress ratio | $P_m + P_b + Q$ stress, (B)x(C) (psi) | Alternating stress, Salt (psi) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | [[ ]] psi | | 122 | 876 | 1.000 | 0.500 | 4.00 | 488 | 20588 | N/A | 3504 | N/A |
| 2 | Level A | 1 | 270 | 1809 | 1.000 | 0.500 | 4.00 | 1080 | 20588 | 0.052 | 7236 | 13025 |
| 3 | Level B | 1 | 584 | 3673 | 1.000 | 0.500 | 4.00 | 2336 | 20588 | 0.113 | 14692 | 26446 |
| 4 | Level B | 2 | 600 | 3873 | 1.000 | 0.500 | 4.00 | 2400 | 20588 | 0.117 | 15492 | 27886 |
| 5 | Level B | 3 | 504 | 4947 | 1.000 | 0.500 | 4.00 | 2016 | 20588 | 0.098 | 19788 | 35618 |
| 6 | Level B | 4 | 506 | 4746 | 1.000 | 0.500 | 4.00 | 2024 | 20588 | 0.098 | 18984 | 34171 |
| 7 | Level B | 5 | 307 | 1105 | 1.000 | 0.500 | 4.00 | 2024 | 20588 | 0.098 | 4420 | 7956 |
| 8 | Level B | 6 | 282 | 1909 | 1.000 | 0.500 | 4.00 | 1228 | 20588 | 0.060 | 7636 | 13745 |
| 9 | Level D | 1 | 800 | 5843 | 1.000 | 0.500 | 4.00 | 1128 | 49410 | 0.023 | 23372 | 42070 |
| 10 | Level D | 2 | 833 | 6165 | 1.000 | 0.500 | 4.00 | 3200 | 49410 | 0.065 | 24660 | 44388 |

NON-PROPRIETARY INFORMATION

## Table EMEB-B-1-16
## Long Gussets Welds ASME Primary and Secondary Stresses

| Item | Service Level | Load Case | (A) Local membrane stress (psi) | (B) Surface maximum stress (psi) | Plate thickness (inch) | Fillet weld size (inch) | (C) Undersized Weld stress factor | $P_m+P_b$ (A) x (C) at weld (psi) | Local membrane Allowable stress (psi) | (D) Primary stress ratio | $P_m + P_b + Q$ stress, (B)x(C) (psi) | Alternating stress, Salt (psi) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | [[ ]] psi | | 1738 | 1740 | 0.500 | 2x0.375 | 1.00 | 1738 | 20588 | N/A | 1740 | N/A |
| 2 | Level A | 1 | 4509 | 4519 | 0.500 | 2x0.375 | 1.00 | 4509 | 20588 | 0.219 | 4519 | 8134 |
| 3 | Level B | 1 | 9040 | 9050 | 0.500 | 2x0.375 | 1.00 | 9040 | 20588 | 0.439 | 9050 | 16290 |
| 4 | Level B | 2 | 9505 | 9515 | 0.500 | 2x0.375 | 1.00 | 9505 | 20588 | 0.462 | 9515 | 17127 |
| 5 | Level B | 3 | 13921 | 13931 | 0.500 | 2x0.375 | 1.00 | 13921 | 20588 | 0.676 | 13931 | 25076 |
| 6 | Level B | 4 | 13455 | 13465 | 0.500 | 2x0.375 | 1.00 | 13455 | 20588 | 0.654 | 13465 | 24237 |
| 7 | Level B | 5 | 5146 | 5156 | 0.500 | 2x0.375 | 1.00 | 5146 | 20588 | 0.250 | 5156 | 9281 |
| 8 | Level B | 6 | 4711 | 4721 | 0.500 | 2x0.375 | 1.00 | 4711 | 20588 | 0.229 | 4721 | 8498 |
| 9 | Level D | 1 | 11598 | 11608 | 0.500 | 2x0.375 | 1.00 | 11598 | 49410 | 0.235 | 11608 | 20894 |
| 10 | Level D | 2 | 12336 | 12346 | 0.500 | 2x0.375 | 1.00 | 12336 | 49410 | 0.250 | 12346 | 22223 |

NON-PROPRIETARY INFORMATION

## Table EMEB-B-1-17
## Original Outer Side Hood ASME Primary and Secondary Stresses

| Item | Service Level | Load Case | (A) Local membrane stress (psi) | (B) Surface maximum stress (psi) | Plate thickness (inch) | Fillet weld size (inch) | (C) Undersized Weld stress factor | $P_m+P_b$ (A) x (C) at weld (psi) | Local membrane Allowable stress (psi) | (D) Primary stress ratio | $P_m + P_b + Q$ stress, (B)x(C) (psi) | Alternating stress, Salt (psi) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | [[ ]] psi | | 691 | 1080 | 0.500 | 0.500 | 1.00 | 691 | 20588 | N/A | 1080 | N/A |
| 2 | Level A | 1 | 1796 | 2542 | 0.500 | 0.500 | 1.00 | 1796 | 20588 | 0.087 | 2542 | 4576 |
| 3 | Level B | 1 | 3577 | 5433 | 0.500 | 0.500 | 1.00 | 3577 | 20588 | 0.174 | 5433 | 9779 |
| 4 | Level B | 2 | 3746 | 5693 | 0.500 | 0.500 | 1.00 | 3746 | 20588 | 0.182 | 5693 | 10247 |
| 5 | Level B | 3 | 3499 | 5936 | 0.500 | 0.500 | 1.00 | 3499 | 20588 | 0.170 | 5936 | 10685 |
| 6 | Level B | 4 | 3307 | 5851 | 0.500 | 0.500 | 1.00 | 3307 | 20588 | 0.161 | 5851 | 10532 |
| 7 | Level B | 5 | 1497 | 3574 | 0.500 | 0.500 | 1.00 | 1497 | 20588 | 0.073 | 3574 | 6433 |
| 8 | Level B | 6 | 1890 | 2773 | 0.500 | 0.500 | 1.00 | 1890 | 20588 | 0.092 | 2773 | 4991 |
| 9 | Level D | 1 | 4247 | 5812 | 0.500 | 0.500 | 1.00 | 4247 | 49410 | 0.086 | 5812 | 10462 |
| 10 | Level D | 2 | 4488 | 6022 | 0.500 | 0.500 | 1.00 | 4488 | 49410 | 0.091 | 6022 | 10840 |

References:

1. Entergy Letter (BVY 04-009) to NRC dated January 31, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No 263, Supplement No. 4, Extended Power Uprate – NRC Acceptance Review" Attachment 7.
2. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Subsection NG, 1989 Edition with no Addenda.
3. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Appendix F, 1989 Edition with no Addenda.

NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-1**
**VYNPS Originally Proposed Steam Dryer Modification**

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-1** (continued)
**VYNPS Originally Proposed Steam Dryer Modification**

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-1** (continued)
**VYNPS Originally Proposed Steam Dryer Modification**

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**Figure EMEB-B-1-2**
**VYNPS Installed Steam Dryer Modification**

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-2** (continued)
**VYNPS Installed Steam Dryer Modification**

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-2** (continued)
**VYNPS Installed Steam Dryer Modification**

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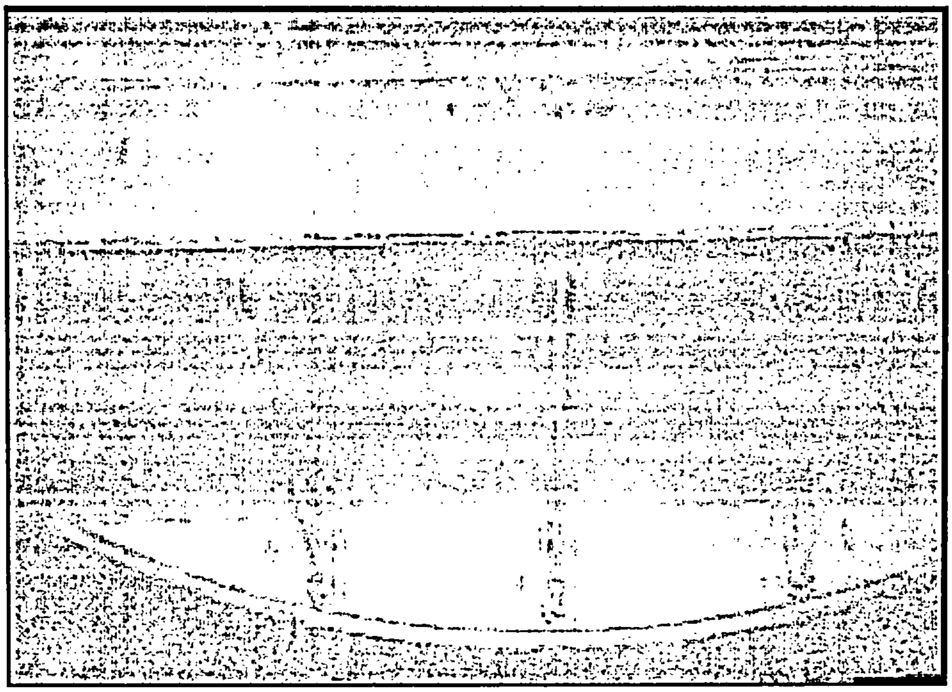
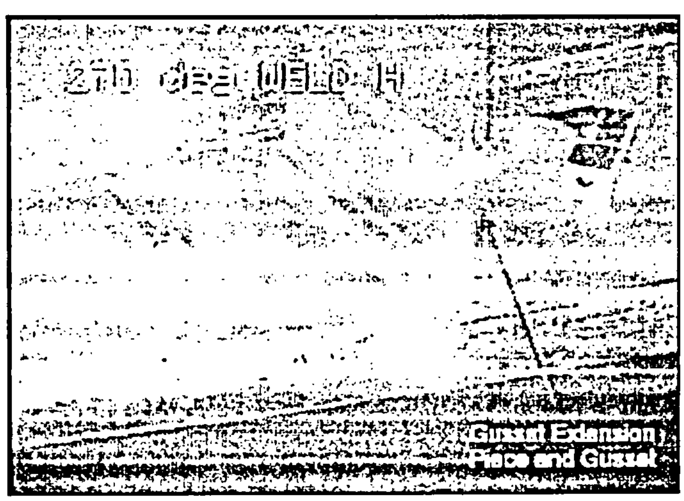
NON-PROPRIETARY INFORMATION

## Figure EMEB-B-1-2a
## VYNPS Steam Dryer Installation Photographs



View of Modified VYNPS Steam Dryer 270 degree side




Detail of Gusset Extension attached to Lower Cover Plate and Hood Gusset

NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-3**
**Steam Dryer Fluctuating Loads – Reference Load Scaling to VYNPS CLTP**

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-4**
**Steam Dryer Fluctuating Loads – Reference Load Scaling to VYNPS EPU**

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-5**
**VYNPS CLTP Response Spectrum**
**Based on Domestic Plant "A" Startup Test Data**

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-6**
**VYNPS CLTP Response Spectrum Based on Foreign Plant A Startup Test Data**

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-7**
**VYNPS CLTP Response Spectrum Based on Foreign Plant B Startup Test Data**

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NON-PROPRIETARY INFORMATION

## Figure EMEB-B-1-8
## VYNPS CLTP Enveloping Response Spectrum

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-9**
**VYNPS CPPU Response Spectrum Based on Domestic Plant "A" Startup Test Data**

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-10**
**VYNPS CPPU Response Spectrum Based on Foreign Plant "A" Startup Test Data**

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NON-PROPRIETARY INFORMATION

## Figure EMEB-B-1-11
## VYNPS CPPU Response Spectrum Based on Foreign Plant "B" Startup Test Data

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-12**
**VYNPS CPPU Enveloping Response Spectrum**

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NON-PROPRIETARY INFORMATION

## Figure EMEB-B-1-13
## Typical TSV Load Time Histories

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NON-PROPRIETARY INFORMATION

## Figure EMEB-B-1-14
## VYNPS Unmodified Dryer

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NON-PROPRIETARY INFORMATION

## Figure EMEB-B-1-15
## VYNPS Modified Steam Dryer

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NON-PROPRIETARY INFORMATION

**Figure EMEB-B-1-16**
**VYNPS Modified Steam Dryer - Details**

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## RAI EMEB-B-2

On April 21, 2004, Entergy briefed the NRC in a conference call with regard to the results of the steam dryer inspection performed during the Spring 2004 outage. Since the steam dryer inspection provides extensive baseline information, the licensee should discuss in detail the cause of each identified crack indication and docket the results of the inspection, including justification of why the results are acceptable with respect to the proposed EPU. It is noted that Quad Cities (QC) did not have a notable dryer failure until after operating at the EPU power level. QC2 has the square type of steam dryer with the internal brace at the outer hood which has failed three times since operating at the EPU power level. The steam dryer at VYNPS is the same design as those at QC. Describe the validation of the steam dryer analysis at VYNPS in successfully predicting steam dryer cracking identified during the Spring 2004 outage.

## Response to RAI EMEB-B-2

During the Spring 2004 outage inspection of the VYNPS steam dryer, four inspection notification reports were generated. None of these reported indications at VYNPS were found in the equivalent locations that generated steam dryer failures at other plants with BWR-3 style steam dryers with internal bracing.

### Inspection Report VYR24-04-01

Indications were found at diametrically opposed locations on the exterior steam dam. Figure EMEB-B-2-1 is a schematic of the locations with respect to the overall dryer.

The first indication was found in welds OP-V19-180 and V02-270. These welds are located at the 215-degree azimuth of the dryer, behind lifting lug "C". Figure EMEB-B-2-2 shows the indication along the top region of the fillet welds joining plate HDE-PL3 and dryer Bank "D". The indication started near the top of V02-270, continued over the top and around the end of the HDE-PL3 plate and proceeded into weld OP-V19-180. The total length of the indication was approximately 3-inches long.

The second indication was found in welds OP-V19-0 and V02-90. These welds are located at the 35-degree azimuth of the dryer, behind lifting lug "A" (diametrically opposite the first indication described above). Figure EMEB-B-2-3 shows the indication along the top region of the fillet welds joining plate HBE-PL2 and dryer Bank "B". The indication started near the top of V02-90, continued over the top and around the end of the HBE-PL2 plate and proceeded into weld OP-V19-0. The total length of the indication was approximately 3-inches long.

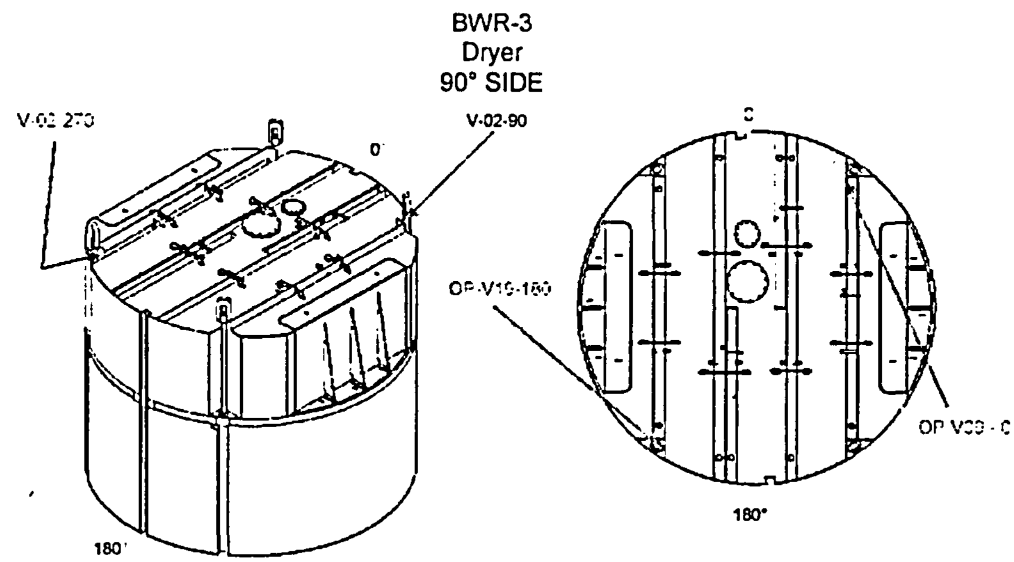
NON-PROPRIETARY INFORMATION

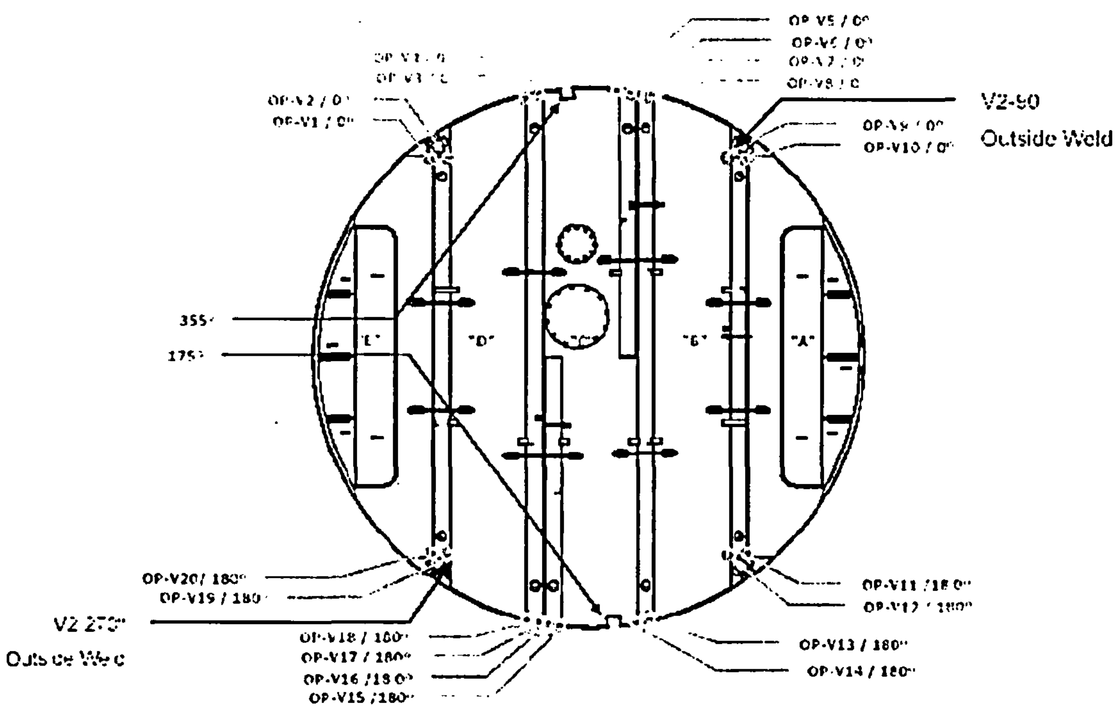
It is believed that the crack initiation was due to high residual stresses generated during the dryer fabrication process. The analysis of the VYNPS steam dryer did not predict these locations as highly susceptible to fatigue cracking. The two crack locations are diametrically opposite each other and very similar in configuration and length. Two other locations in the steam dryer (144 degrees and 324 degrees) are structurally identical and see the same loading conditions; these locations have no observed evidence of cracking. This points to the likelihood of the presence of an additional contributing factor aside from the pressure loads during normal operation. The VYNPS steam dryer was made in halves and the two cracks are in identical locations in each half. This evidence indicates that a high residual stress condition was probably developed by the original dryer fabrication welding sequence; as the fabrication sequence progressed from one side of the dryer half to the other, there would be an asymmetrical distribution of the residual stresses within the dryer half. Other "cold spring" type loading could also have been generated during the fabrication process. A high mean stress (from weld residual stress) would have significantly reduced the vibration load required to initiate the crack. After the crack developed, the residual stress would have been relieved and the crack growth subsided. This is consistent with the cracks being very similar in length.

The repair of the cracked locations consisted of grinding out the locations to one inch beyond the crack tip and reapplying a ½ - weld. The grind out of the crack and weld repair at the V02-90 location is shown in Figure EMEB-B-2-4. In addition, a repair fixture was installed at both the observed crack locations (35-degree and 215-degree azimuths) and the structurally identical locations (144 degrees and 324 degrees) to provide additional confidence that no additional cracking will occur at these locations during operation at CPPU conditions. Each repair fixture was welded between dryer band top hood and the steam dam plate where the cracks were observed.

NON-PROPRIETARY INFORMATION

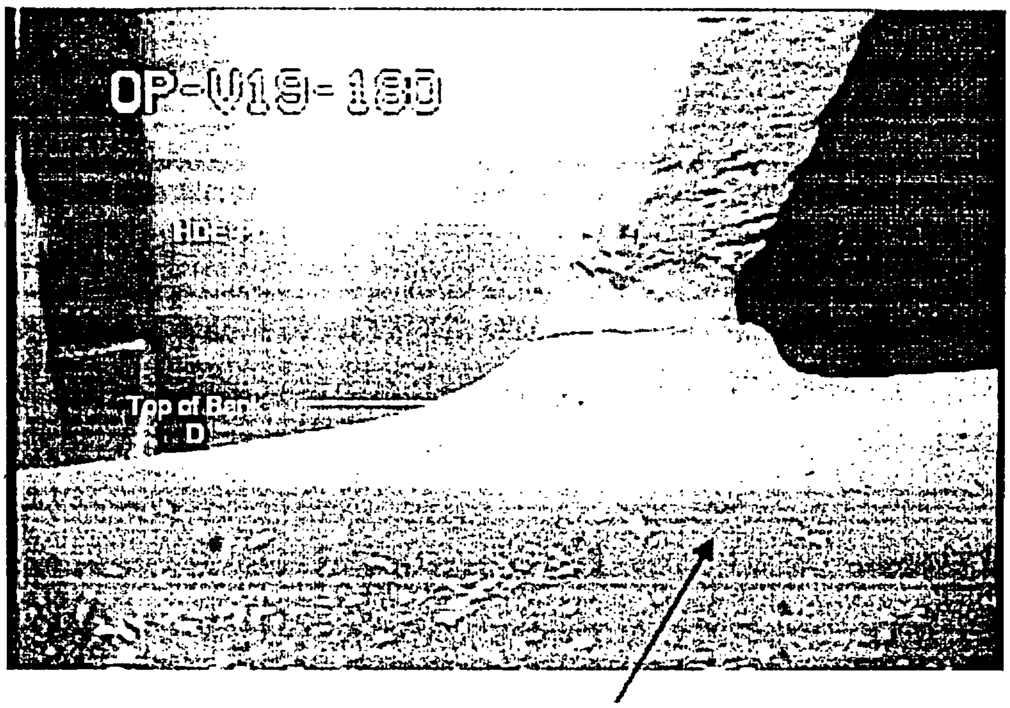
Figure EMEB-B-2-1
VYR24-04-01 Indication Locations

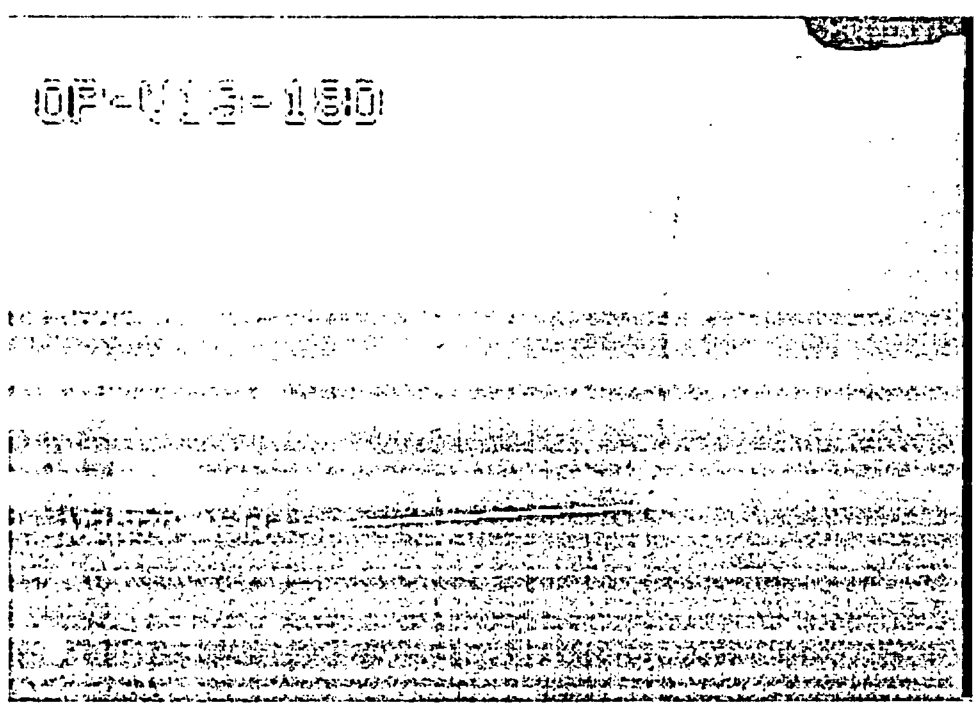


NON-PROPRIETARY INFORMATION

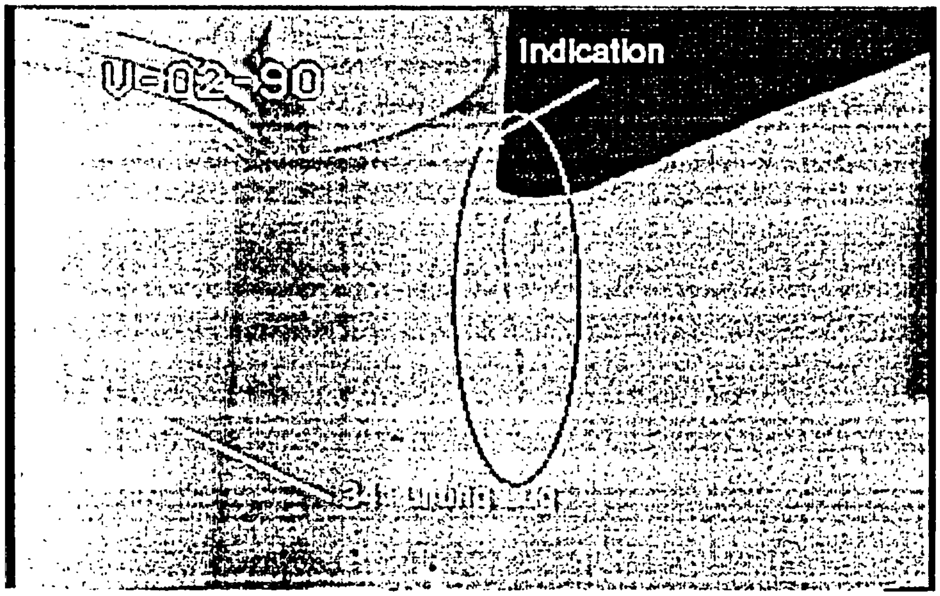
Figure EMEB-B-2-2
Inspection Photographs of 215-degree Azimuth Indication

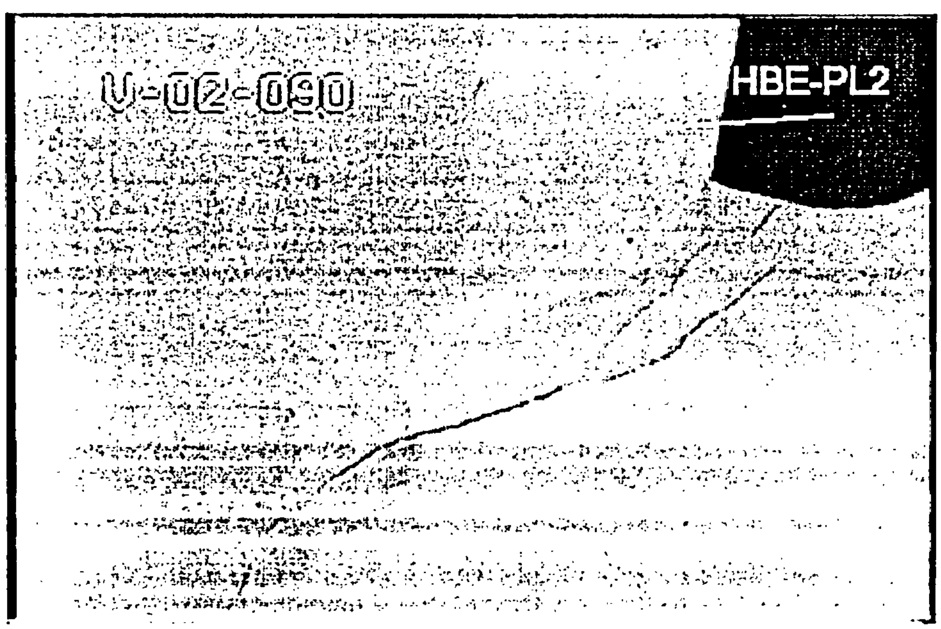


NON-PROPRIETARY INFORMATION

**Figure EMEB-B-2-3**
**Inspection Photographs of 35-degree Azimuth Indication**

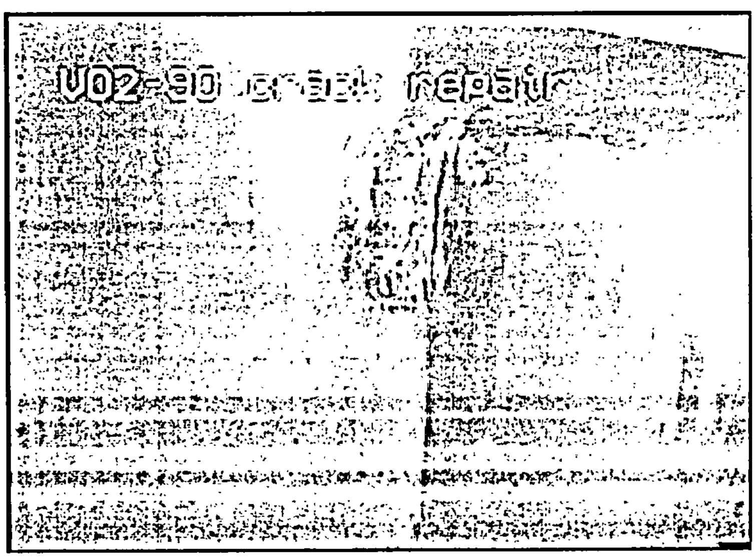


NON-PROPRIETARY INFORMATION

**Figure EMEB-B-2-4**
**VYR24-04-01 Crack Grind Out and Weld Repair**

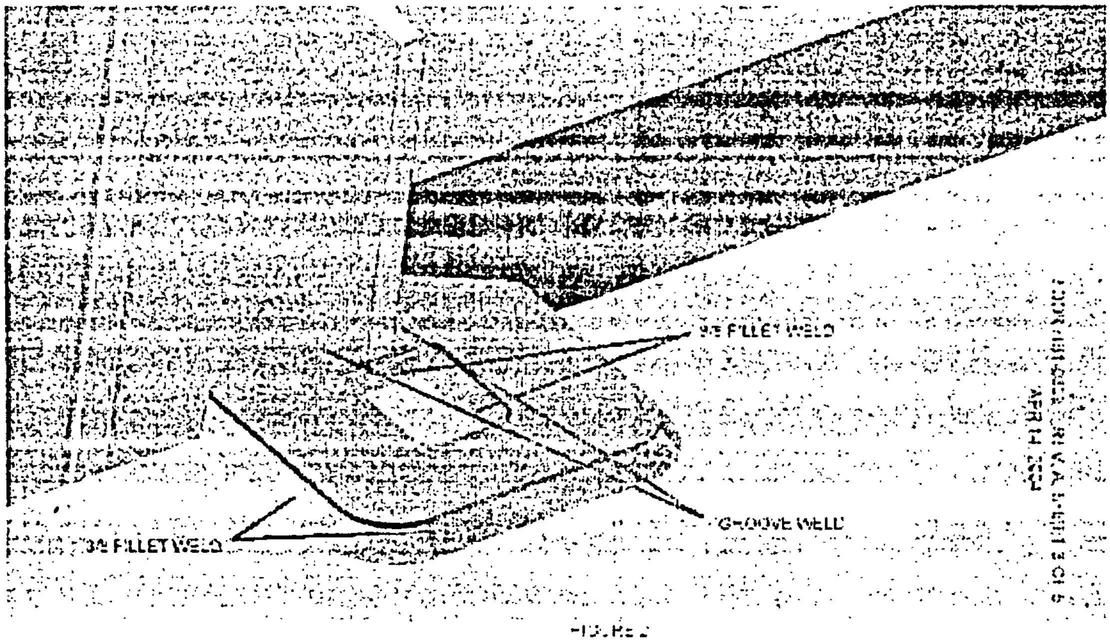

Weld Grind Out

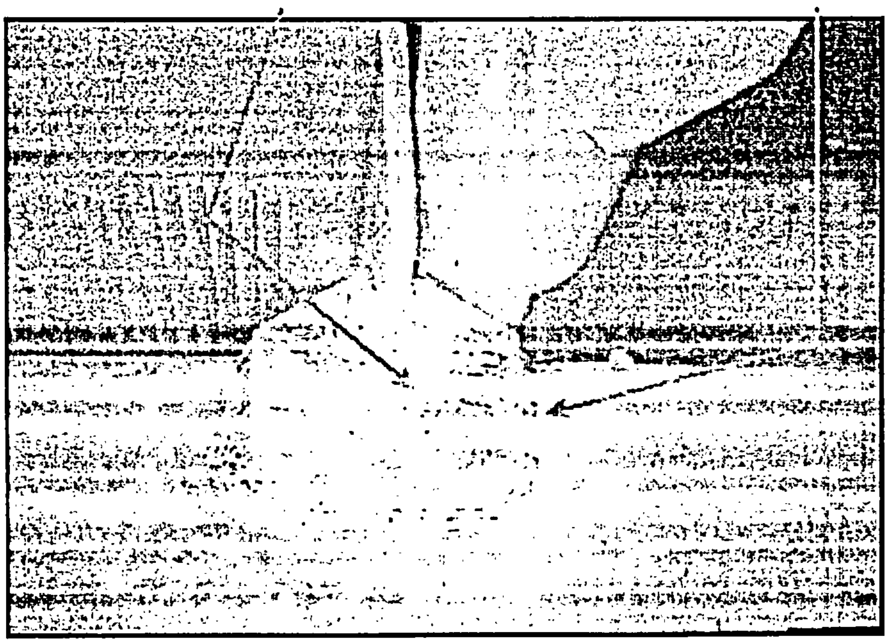


Weld Repair

NON-PROPRIETARY INFORMATION

**Figure EMEB-B-2-4 (cont)**
**Steam Dam Repair Fixture**



Field Installation of Repair Fixture

During the inspection of the VYNPS steam dryer interior surfaces using a remote operated vehicle, indications were observed at various dryer bank end plates.

Figure EMEB-B-2-5 is a schematic of the end plate locations with respect to the overall dryer and shows a representative photograph of the indications. Table EMEB-B-2-1 summarizes the location and number of indications found on each interior end plate:

**Table EMEB-B-2-1**
**VYNPS Dryer Inspection End Plate Indications Summary**

| Bank | Location | Description | Number of Indications |
|---|---|---|---|
| B | 0° end of bank adjacent to Weld V4 | Horizontal indications in unit end plate base material | 7 |
| D | 180° end of bank adjacent to Weld V1 | Horizontal indication in unit end plate base material adjacent to unit end plate to trough weld | 1 |
| C | Near center of bank adjacent to Weld V6-0 | Horizontal indication in unit end plate base material | 6 |
| C | Near center of bank adjacent to Weld V6-180 | Horizontal indication in unit end plate base material | 2 |

Each VYNPS steam dryer vane module assembly (dryer unit) includes two end plates. There are 16 total dryer units in the VYNPS dryer assembly. Unit end plates have a channel cross-section 8.75 inch wide with 1.25-inch high flanges and are formed from 3/16-inch thick Type 304 stainless steel. The bottom end of each unit end plate has two notches 3 inches high by ½ or 1/4-inch deep (removing most of the flanges). The upper end of each end plate has similar notches except the inlet side notch is 1-inch high and the outlet side notch is 1.5-inches high. Figure EMEB-B-2-6, taken from the vendor's dryer unit fabrication drawing shows these details. After notching, the width of the lower 3 inches of the dryer unit end plate is 8 inches. The notched lower end of the dryer unit end plate is fit between the vertical sides of the lower trough of the dryer bank (an 8-inch wide space). Similarly, the notched upper end of the dryer unit end plate fits between ½-inch thick vertical rail pieces as shown in Figure EMEB-B-2-7.

Each dryer bank includes three or four dryer units that are installed into the bank trough. 3/16-inch vertical fillet welds are used to join the end plates of adjacent dryer units. Bank C is split into two halves with opposite steam flow directions. At both ends of the bank, the last dryer unit end plate is welded to the ½-inch thick cylindrical shell plate with a 3/16-inch fillet weld. On the outlet side, a ¼-inch thick closing piece is used between the unit end plate and the ½-inch vertical shell plate (see Figure EMEB-B-2-8 for typical details).

After installation, the weight of the dryer units, about 20 lbs per inch of bank length, is transferred to 1/2-inch by 1-inch bars that are welded 3 inches below the top edge of the trough along the full length of the bank as shown in Figure EMEB-B-2-7. Lateral support for the lower end of the dryer units is provided by the vertical 1/2–inch thick trough members and plates at the end of each trough. The upper rail pieces, as shown in Figure EMEB-B-2-7, capture the upper end of the dryer units. Tie rods, nuts and spacers control spacing of the dryer vanes themselves as shown in Figure EMEB-B-2-6.

The VYR24-04-02 inspection report documents a total of 16 indications located on the 1.25-inch wide flange of four end plates (Refer to Table 1 for indication locations). The indications are at random locations on the end plates and exhibit different characteristics. Some indications appear jagged and others straight. The indications are not in close proximity to each other, but are scattered randomly. Some indications are across the full 1.25-inch flange width and others are not. One indication is located in the heat affected zone (HAZ) of the fillet weld joining the flange to the bottom trough, and is clearly IGSCC. The indications are very tight suggesting they are shallow. Two of the indications may be surface discontinuities and not actual cracks. The indications are only on inlet side flanges (the outlet sides were also examined). The indications do not go into the vertical welds. It is not possible to determine if any of the indications are thru-wall.

IGSCC appears to be the cause for some of the indications based on jagged appearance and location in weld HAZ material. The dryer unit end plates are located in the dryer interior and are not subjected to any main steam line acoustic loading. IGSCC in steam dryers has been typically limited in depth and length since in many cases it is caused by cold work or weld induced residual stress. In many cases, flaws in steam dryers have appeared but eventually slow down/arrest because the flaws grow through or away from the localized areas of residual stress.

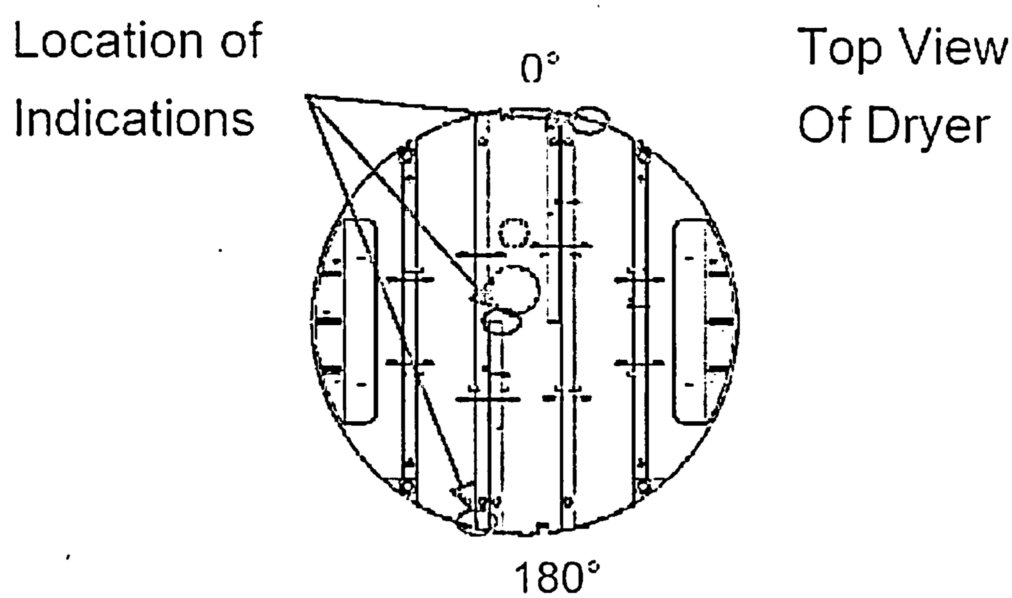
The dryer unit end plates, with indications, are securely attached and captured within the structure of the steam dryer bank assembly. The vertical edges of these end plates are attached to the dryer assembly with 3/16-inch fillet welds (each weld approximately 48-inches long). There were no relevant indications reported in these vertical welds. The geometric configuration of unit end plates is such that the upper and lower edges are mechanically captured by the steam dryer assembly as shown in Figure EMEB-B-2-9. The reported horizontal indications were seen in the 1.25-inch inlet side end plate flange. The vanes prevent inspection of the central end plate surface, but inspection of the outlet side end plate flanges found no indications.

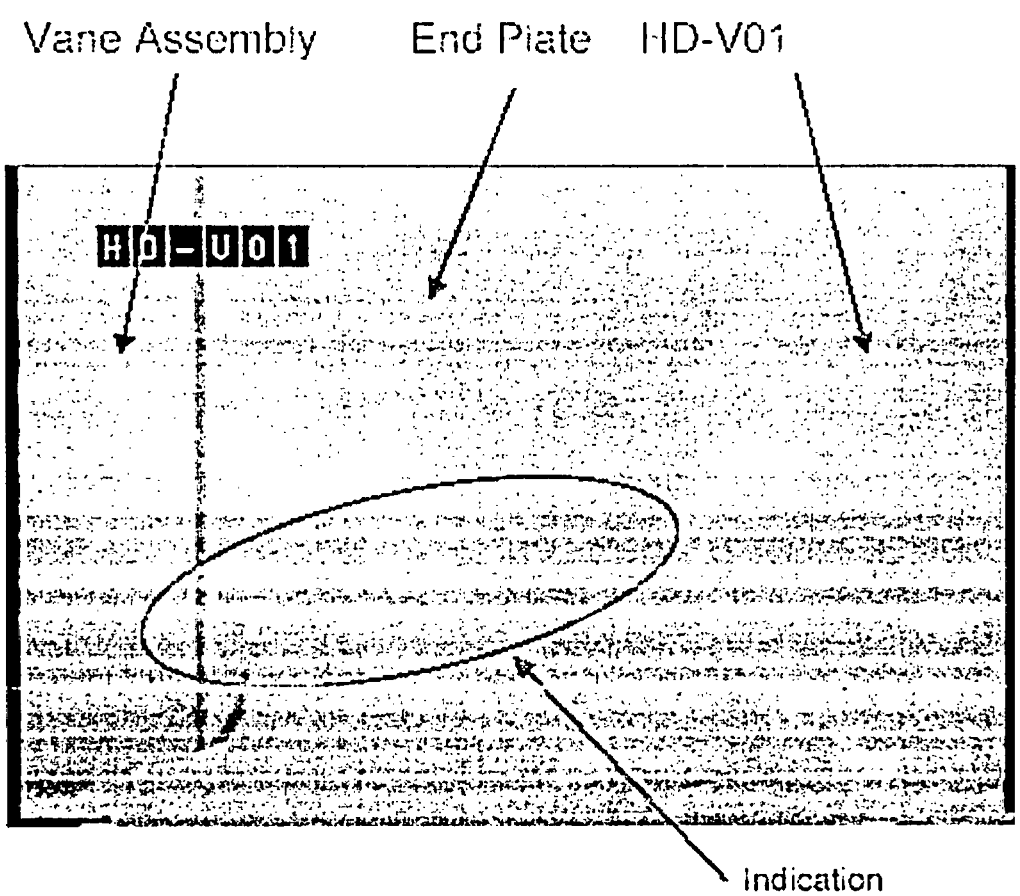
A worst-case scenario would postulate that the indications could propagate from the inlet side flange across entire end plate including the outlet side flange. This is very unlikely, but it would not result in any significant structural or performance impact to the steam dryer. For the purpose of this discussion it is postulated that the end plate horizontal indications propagate across the entire 8.75-inch unit end plate width including both the inlet and outlet side flange, as shown in Figure EMEB-B-2-9. Such full width through thickness cracks would have no structural impact nor is there any concern for loose parts. The separated end plate sections, as shown in Figure EMEB-B-2-9, are still attached and will continue to function.

The steam dryer analysis for VYNPS did not predict steam dryer cracking due to fatigue in the dryer bank end plates.

NON-PROPRIETARY INFORMATION

Figure EMEB-B-2-5
VYR24-04-02 Indication Locations

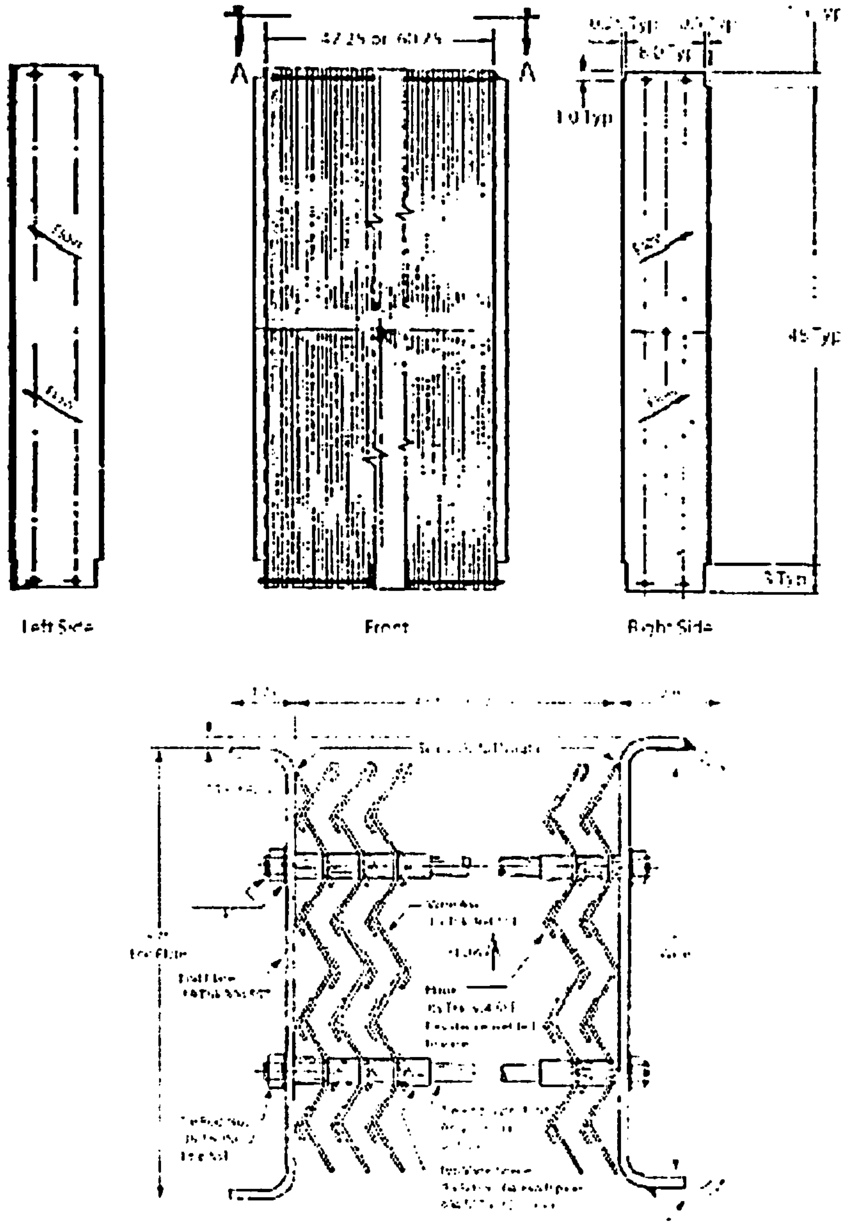



Representative Indication on End Plate

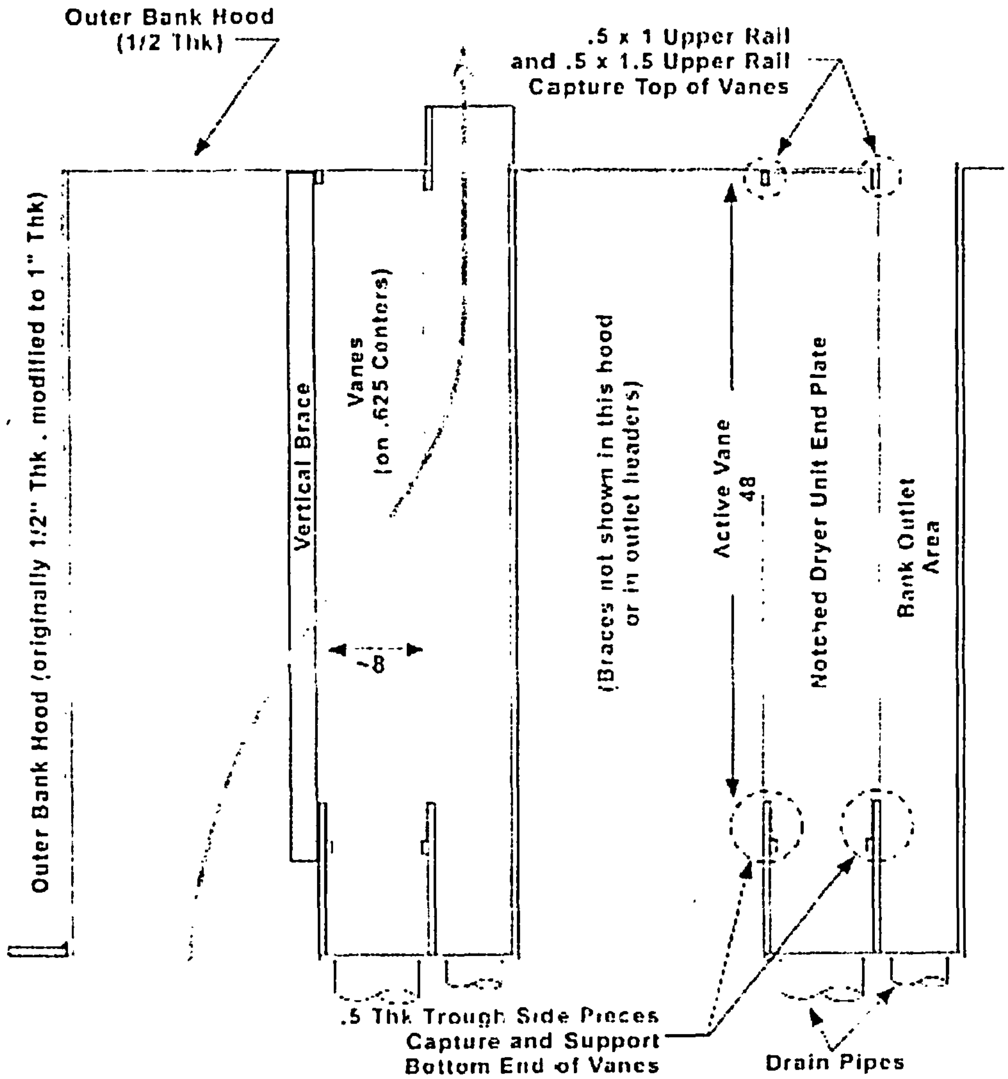
NON-PROPRIETARY INFORMATION

Figure EMEB-B-2-6
Typical VYNPS Steam Dryer Unit



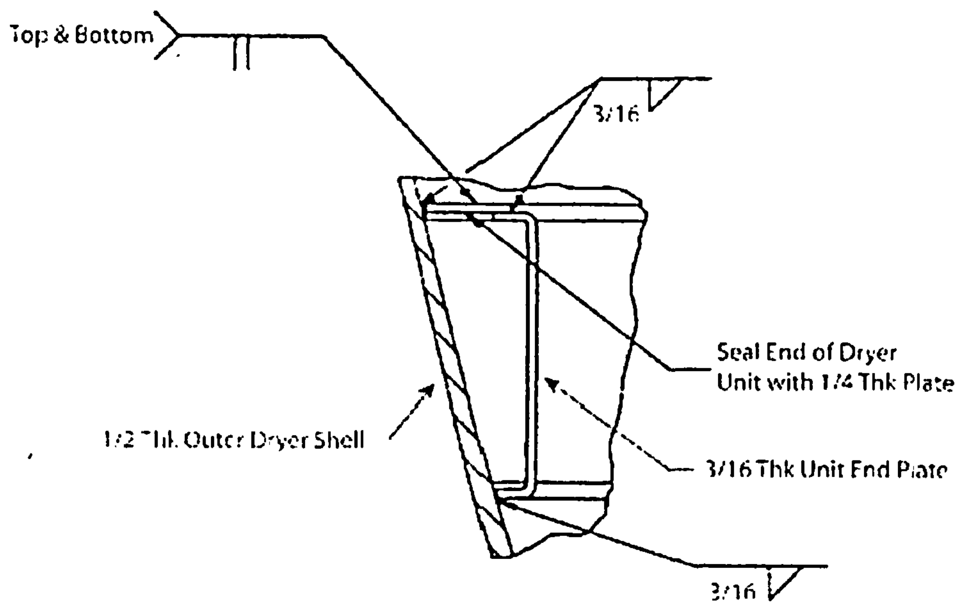
NON-PROPRIETARY INFORMATION

**Figure EMEB-B-2-7**
**Cross-Section Through Banks Showing Vane Capturing Features**



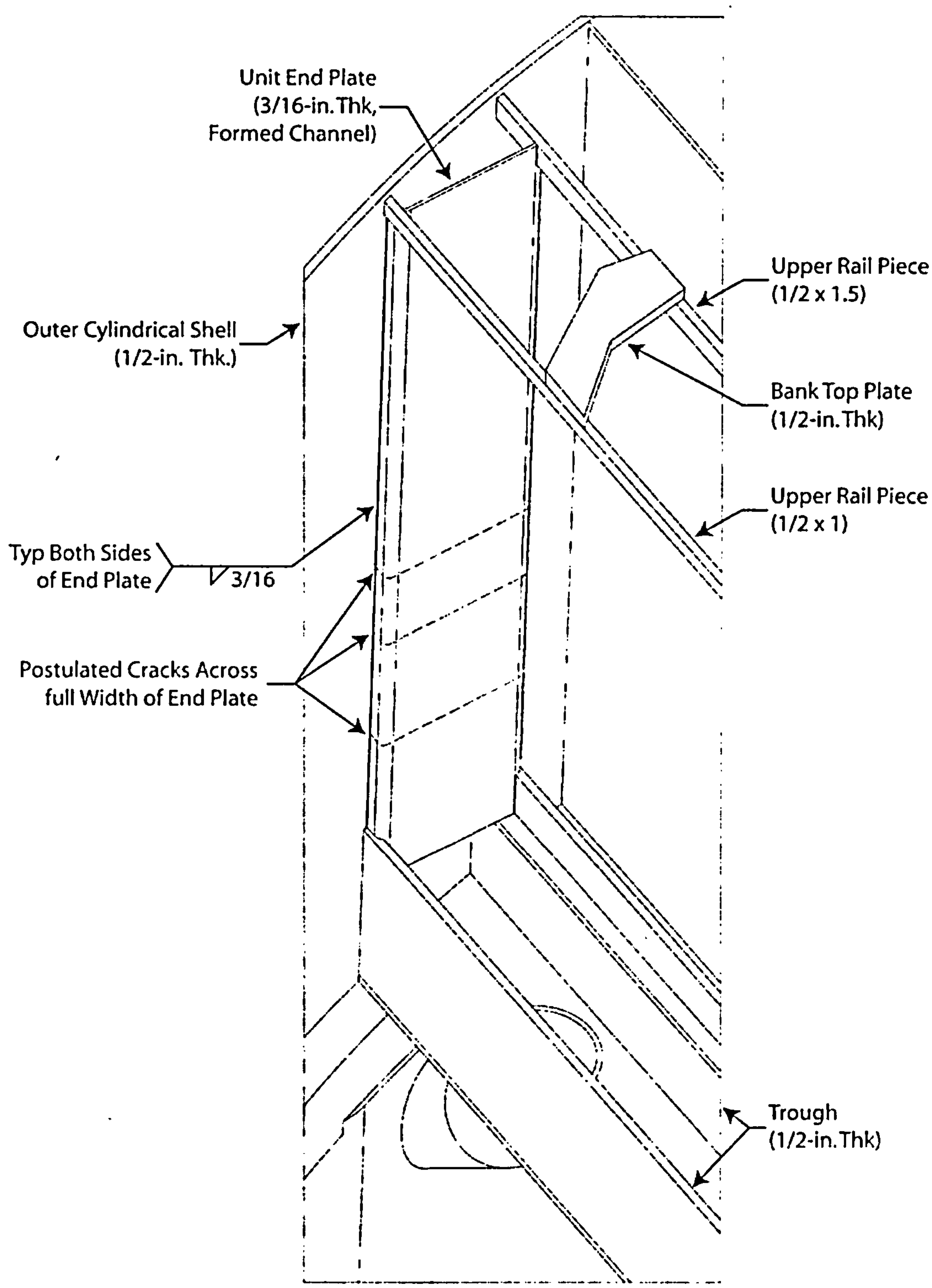
NON-PROPRIETARY INFORMATION

Figure EMEB-B-2-8
Connection of Unit End Plate with Dryer Assembly



NON-PROPRIETARY INFORMATION

Figure EMEB-B-2-9
Cut-away of Bank Showing Unit End Plate



Inspection Report VYR24-04-03

Inspection of the VYNPS steam dryer interior surfaces using a remote operated vehicle indicated regions of the dryer internal plates with crud buildup. Upon detailed review of the inspection tapes, these observations were dispositioned by GE engineering as having no effect on the structural integrity of the dryer or any other RPV internals.

Inspection Report VYR24-04-04

During the internal inspection in the Drain Channel DC-V04C weld, an indication was noted starting near the top of the weld and continuing down approximately 12.0 inches. Additionally, an indication in the drain channel adjacent to the drain pipe at 185 degree azimuth was noted. Figure EMEB-B-2-10 is a schematic of the indication locations with respect to the overall dryer. Figures EMEB-B-2-11 and 12 are photographs of the indications from the inspections.

Both of these indications are located in the dryer skirt region. The lower skirt region has a history of minor indications in several plants at both original and uprate power levels. Cracks have occurred in the drain channel attachment welds and in the skirt near the drain channels and guide channels. Both IGSCC and high cycle fatigue have been identified as failure mechanisms for these cracks; the cause depends on the circumstances for the individual failure. Fatigue cracking in BWR 4/5/6 drain channel welds have been discussed in GE Service Information Letter SIL 474. Because the lower skirt is partially submerged in the water, the skirt is subject to both the FIV fluctuating pressure loads that act on the upper components of the dryer and hydrodynamic loads from the liquid flow spillover from the steam separators. The fluctuating pressure loads on the skirt will be somewhat attenuated at both CLTP and CPPU conditions by the narrow annular gap between the skirt and the vessel wall and are lower than the pressure loads on the upper components of the steam dryer. The fluctuating pressure loads on the skirt will increase at CPPU conditions. There is no increase in core flow rate with CPPU. At the higher CPPU power levels, the liquid spillover flow will be less. In addition, the water level inside the skirt will be lower at CPPU power levels. It is expected that both of these effects will result in a reduction in the hydrodynamic loads on the skirt. The overall effect of these changes is that the loading on the lower skirt region will not be significantly affected by CPPU.

Drainpipe Indication

The VYNPS steam dryer is constructed from Type 304 stainless steel with no special chemistry controls. BWR experience has shown this material to be susceptible to IGSCC in the BWR steam and water environment. The indication is located in the weld heat affected zone ¼-inch thick drain channel material adjacent to a 3-IPS Sch 40 drainpipe-to-drain channel weld at approximately 185°. This indication is estimated as 3 inches in length.

Based on the appearance and location of the 3-inch long drainpipe indication this is most likely IGSCC. IGSCC in steam dryers has been typically limited in length since in many cases it is caused by cold work or weld induced residual stress. In many cases, flaws in steam dryers have appeared but eventually slow down/arrest because the flaws grow through or away from the localized areas of residual stress. The indication does not appear to be open sufficiently to allow any steam bypass and there is no apparent staining to indicate that liquid is leaking through the crack. The current 3 inch long indication is not expected to increase to more than 4.2 inches in length during the next 18-month cycle based on BWRVIP growth rate of 5x10-5 in/hr that is considered conservative for this case. A 4.2-inch long indication in this location, even if assumed through wall, would not have a significant impact on the function or structural integrity of the drainpipe or drain channel. Further discussion is contained in the response to NRC RAI EMCB-A1.

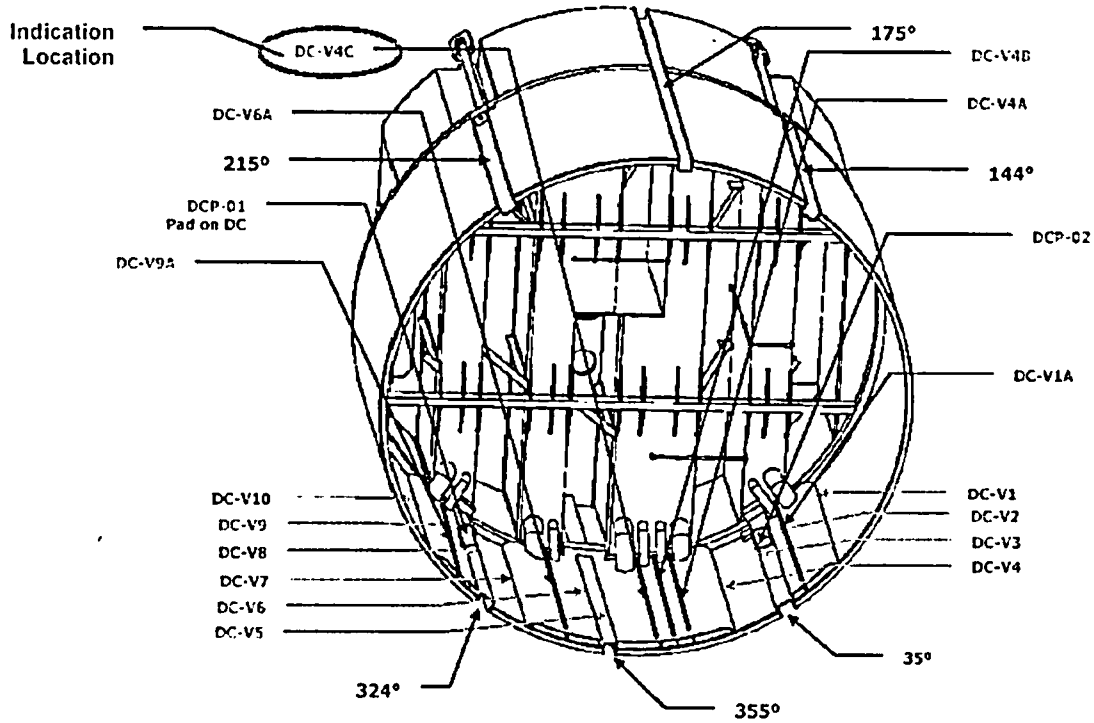
Drain Channel (DC-4VC) Weld

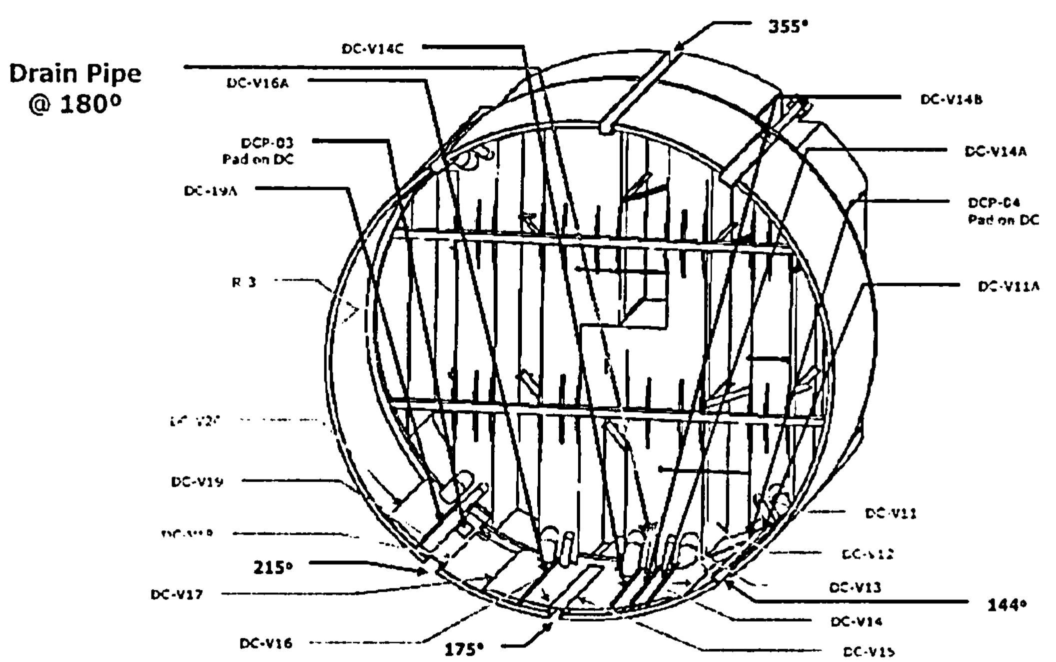
The second indication discussed in inspection report VYR24-04-04 is located to the left side of the 91.5-inch long vertical drain channel (DC-4VC) weld as shown in Figure EMEB-B-2-13. A cross section through this drain channel with a detail for weld DC-4VC is shown as Figure EMEB-B-2-14. Because weld DC-4VC is one of several final field assembly welds, it is possible that this was an area of less than optimum fit up. The length of this indication is estimated at approximately 12 inches or 13% of the weld length. Careful examination with the Remotely Operated Vehicle was able to show that the indication is located in weld HAZ base material and not the weld itself. There is visual evidence of heavy grinding in the area of the indication. Based on the HAZ location and the somewhat jagged appearance this indication is most likely IGSCC.

Because this indication is most likely IGSCC it is expected that the crack is likely to continue to propagate slowly (less than 0.6 inch extension at each end per 18-month cycle based on a reasonable growth rate of 5x10-5 in/hr) and it may become stable. At its current size and location it has no functional impact. The 3-inch and 6-inch drain pipes that are welded to the drain channel section near the crack provide some added structural redundancy to the upper portion of the drain channel sections on either side of the cracked weld (see Figure EMEB-B-2-13). Further discussion is contained in the response to NRC RAI EMCB-A1.

NON-PROPRIETARY INFORMATION

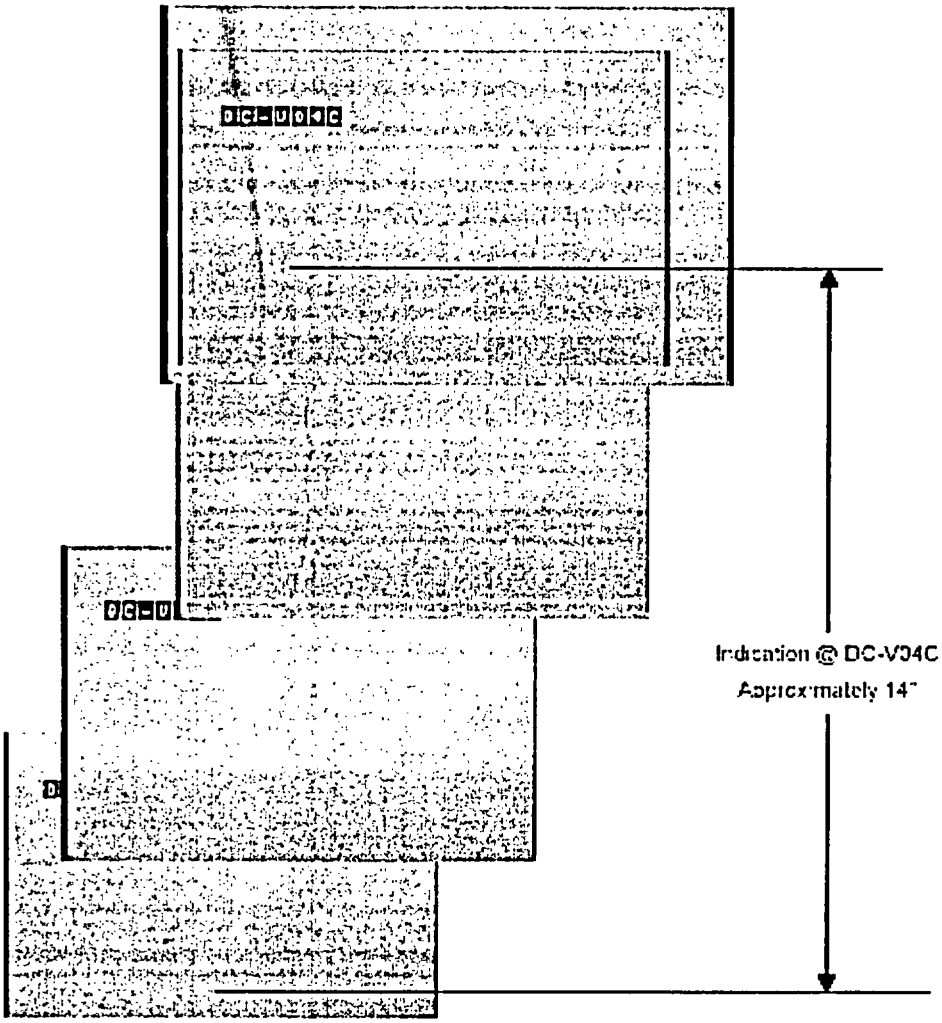
Figure EMEB-B-2-10
VYNPS Dryer Indications Schematic



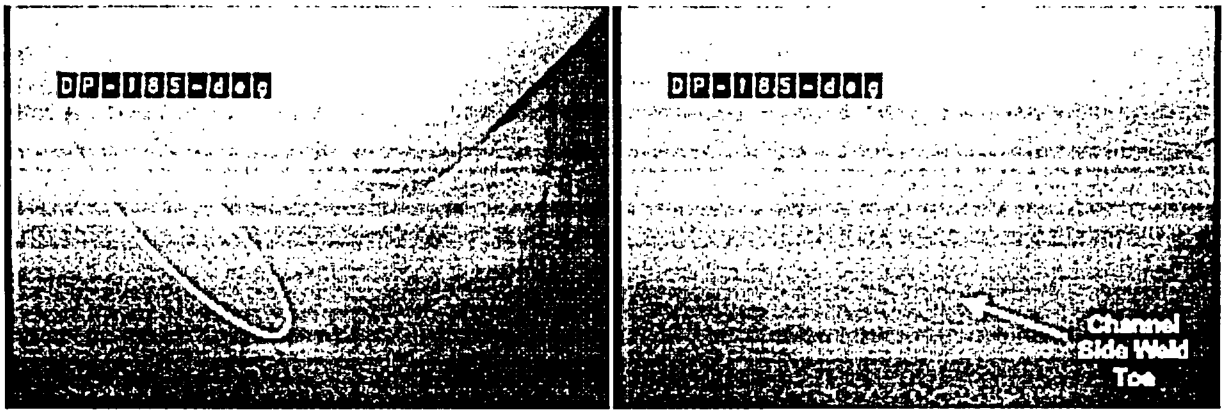
NON-PROPRIETARY INFORMATION

Figure EMEB-B-2-11
Inspection Indications – Drain Channel

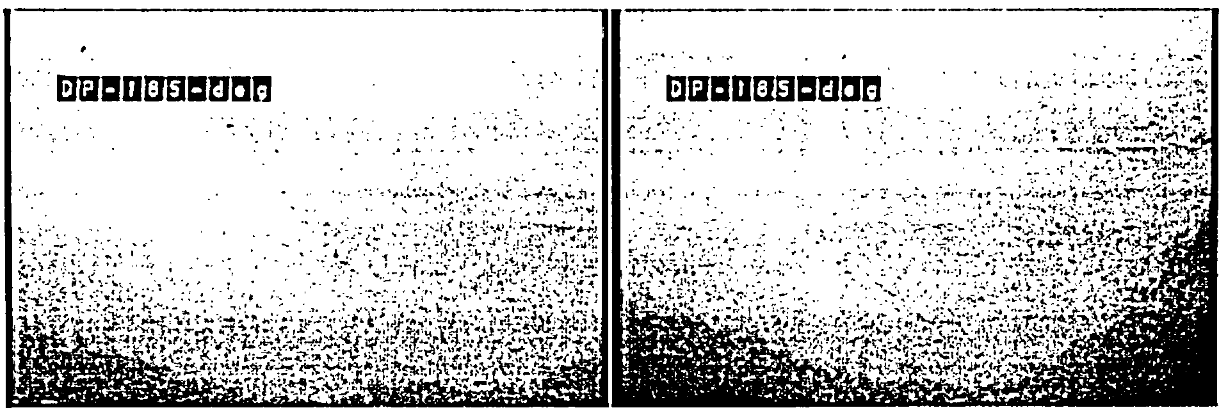


NON-PROPRIETARY INFORMATION

## Figure EMEB-B-2-12
## Inspection Indications – Drain Channel

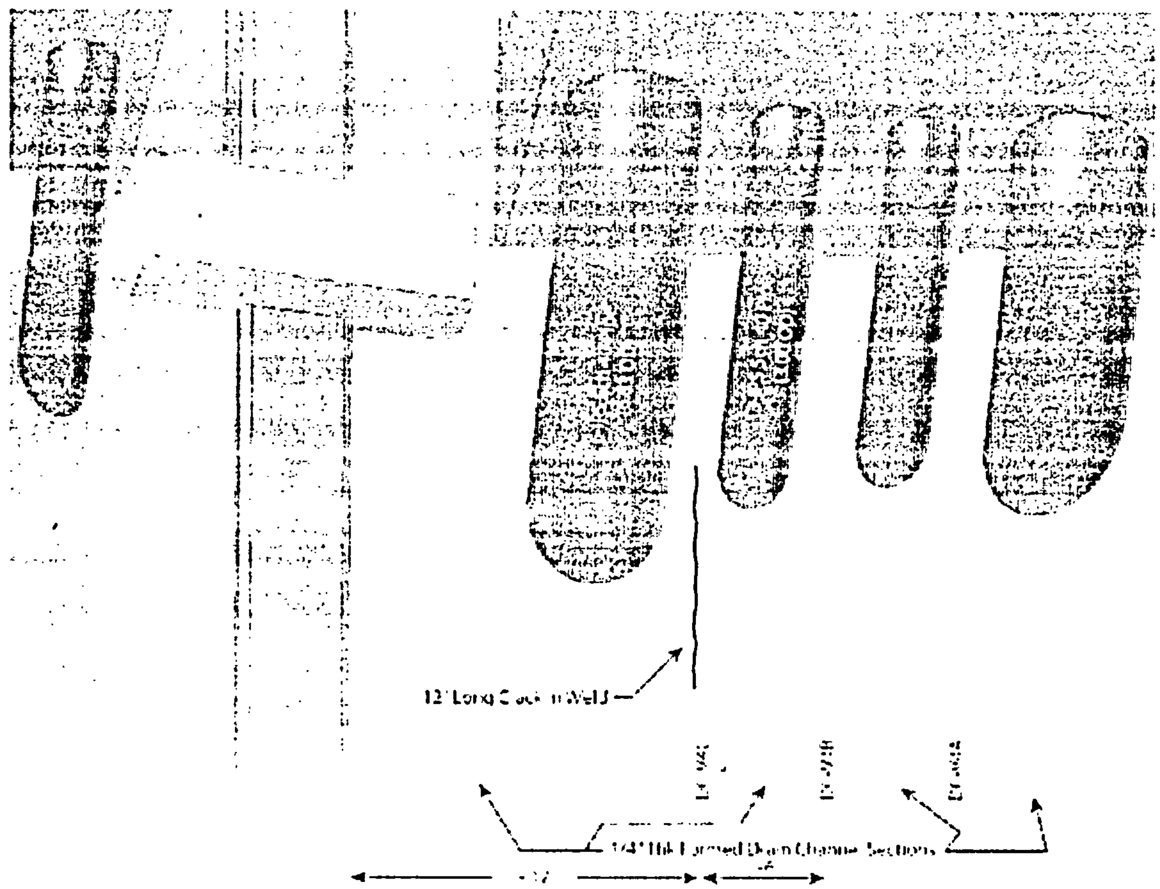



This weld appears to be ground flush and is very difficult to see the weld toe. But the indication appears to be at the drain channel weld toe HAZ area and not in the weld material.



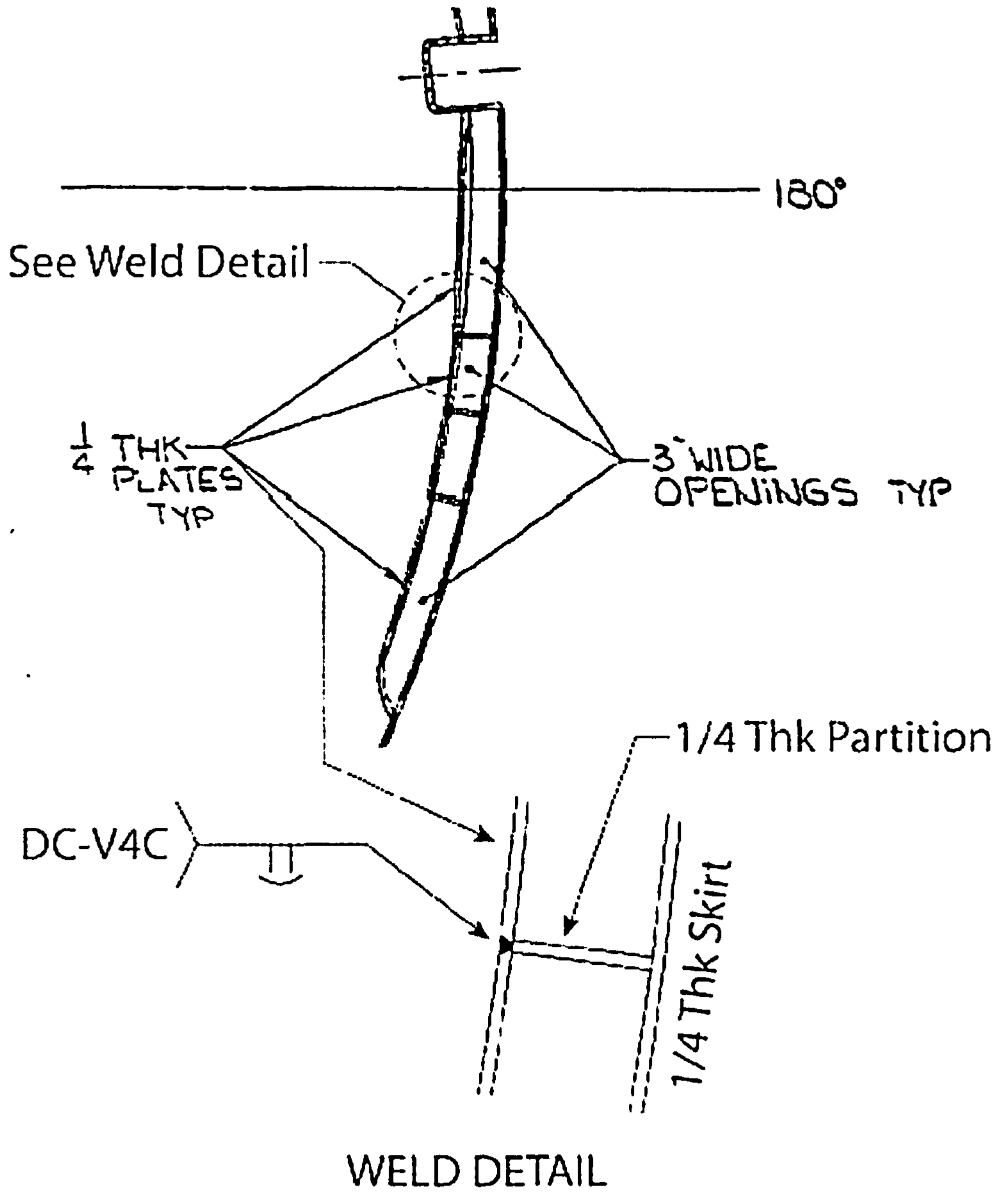
NON-PROPRIETARY INFORMATION

Figure EMEB-B-2-13
VYNPS Drain Channel Indications Schematic



NON-PROPRIETARY INFORMATION

Figure EMEB-B-2-14
VYNPS Drain Channel Cross Section



## RAI EMEB-B-3

Supplement 4 (Reference 5), Attachment 8, page 6, states that VYNPS plant-specific data for the steam dryer pressure loading is not available. Section 4.1 on this page discusses the overall process developed by General Electric (GE) whereby available steam dryer pressure loading data from other plants has been converted into a reference load distribution versus frequency plot that can be further scaled for plant-specific evaluation use. The reference load definition used detailed pressure versus frequency spectrums taken from in-plant measurements recorded for one domestic and two foreign GE boiling water reactor (BWR) plants. As discussed on page 41 of GE report GENE-0000-0018-3359-NP, "Technical Assessment, Quad Cites Unit 2, Steam Dryer Failure - Determination of Root Cause and Extent of Condition," dated August 2003 (ADAMS Accession No. ML032340379), at the domestic plant, the pressure was measured in the middle of the cover plates of the outer bank hood in the 90 degree and 270 degree azimuth. In the two foreign reactors, the pressure sensors were located below the dryer ring, on the skirt and drain channels. For the QC2 event, it was considered more appropriate to use the pressure measurements from the domestic plant since the pressure measurement location was in the region of interest. Based on the lessons-learned from QC2, provide justification for the applicability of the pressure data used for the VYNPS plant-specific application.

## Response to RAI EMEB-B-3

The August 2003 Quad Cities 2 evaluation used only the cover plate sensor pressure data from the domestic plant because the methodology for correlating the pressure measurements at various locations of the dryer had not been developed at the time the QC2 evaluation was performed. The pressure measurements taken at the lower cover plate location were the only measurements that could be applied directly to the hood location that failed at QC2. In order to develop a fluctuating pressure load that modeled all the observed characteristics, the in-plant measurements from all three plant tests were used in developing the load definition used in the VYNPS dryer analysis. As described in the response to EMEB-B RAI 5, the scale model test results for model sensor locations on the outer hood, skirt, and inner banks were used to adjust the in-plant pressure measurements taken at the various locations (e.g., dryer skirt, instrumentation mast) to determine an equivalent pressure at the outer hood. Since the resulting generic load definition modeled all the characteristics observed in the in-plant measurements and includes the measurements from the testing at the domestic plant, this load definition is acceptable for use in the VYNPS dryer analysis.

## RAI EMEB-B-4

Describe the manner in which the steam dryer analyses at VYNPS avoids the weaknesses in the steam dryer analyses applied at QC that lead to the catastrophic failures of the steam dryers at the QC units. Describe the validation of the VYNPS steam dryer analyses to accurately predict the hydrodynamic loading at specific locations of the steam dryer. Describe the structural evaluation of the steam dryer at VYNPS to successfully withstand the hydrodynamic loading under EPU conditions.

## Response to RAI EMEB-B-4

The hydrodynamic loading, structural evaluation, and dryer modifications for the VYNPS steam dryer at EPU conditions are described in Supplement 4 Attachment 7 and in the response to EMEB-B RAI 1. The steam dryer analyses for VYNPS are based on the experience gained from the 2002 and 2003 dryer failures at QC2. The modifications performed on the VYNPS dryer during the Spring 2004 address the structural vulnerabilities identified by the 2002 and 2003 QC2 dryer failures and incorporate the design features for reducing local stress concentrations learned from the 2004 QC2 repairs. Therefore, the modified dryer at VYNPS is designed to successfully withstand the hydrodynamic loading under EPU conditions.

The structural vulnerability that led to the 2002 QC2 lower cover plate failure was the 1/4 inch thick lower horizontal cover plate. It is believed that an acoustic pressure load was acting on the lower cover plate with a frequency that was near the natural frequency of the plate. The thin cover plate also limited the size of the attachment welds, which limited the stresses that could be withstood by the plate. The stresses from the outer vertical hood plate may also have contributed to the failure of the small welds on the lower cover plate. The structural vulnerability that led to the 2003 hood failures in the Quad Cities dryers are the 5" x 7" internal gussets (brackets) that serve as attachment points for the internal diagonal braces. These internal gussets (brackets) cause a very high local stress concentration in the outer hood plates. Fatigue cracks initiated in the outer hood plates at these gusset (bracket) corners during EPU operation in all four dryers at the Dresden and Quad Cities units, with the cracks at Quad Cities 1 and 2 growing to the point of failure. The cracking found during the Spring 2004 outage in the 2003 QC2 dryer repairs resulted from local stress concentrations introduced by the as-installed repair configurations. Minor field modifications were made to the repair configuration during installation. These modifications were judged to be acceptable at the time of installation; however, detailed structural analyses performed after the Spring 2004 cracking showed that the changes introduced a high enough local stress concentration to initiate cracking.

The modifications made to the VYNPS dryer during the Spring 2004 outage eliminated the structural vulnerabilities identified by the Quad Cities failure experience. The lower cover plate was replaced with a thicker plate that raised the natural frequency of the plate above the range of acoustic pressure loads. The thicker cover plate allowed for larger welds to be used, thereby reducing the stresses in these welds. The internal gussets (brackets) in the outer hoods were eliminated with the replacement of the outer vertical hood plates and upper horizontal cover plate. The full height replacement vertical plates (with an increased thickness of one inch) included shop-welded full-length gussets that incorporate the design features for reducing local stress concentrations learned from the 2004 QC2 repairs. These modifications resulted in a significant reduction in the stresses in the VYNPS dryer, making it suitable for EPU operation.

As evidenced by the successful operating histories of the BWR plants with square hood dryers at both original licensed thermal power and EPU conditions, the structural vulnerabilities of the square hood dryer design do not mean that all square hood dryers with internal braces will fail like those at Quad Cities. The fluctuating pressure loads acting on the dryer are believed to be proportional to the steamline flow velocity. The Dresden and Quad Cities steamline flow velocities at both original licensed power and

EPU are the highest in the BWR fleet. It is likely that the loads were high enough to induce the hood failures at Quad Cities only under EPU conditions. The steamline flow velocity for VYNPS at EPU is equal to the steamline flow velocity at Dresden and Quad Cities at original licensed thermal power. These plants operated at these steamline velocities for over 25 years without evidence of outer bank cracking. Therefore, the fluctuating pressure loads acting on the VYNPS dryer at EPU are expected to be substantially lower than the loads that caused the dryer failures at Quad Cities under EPU conditions. Additionally, the dryer at VYNPS was preemptively modified using the same modification design and fabrication methods as employed in the Spring 2004 modifications at Quad Cities to further reduce the fluctuating stresses in the outer cover plate and the outer banks. Based on the conservative structural modifications made to the dryer and using the relatively lower loads acting on the dryer, there is confidence that the modified dryer at VYNPS will successfully withstand the hydrodynamic loading under EPU conditions.

## RAI EMEB-B-5

Supplement 4 (Reference 5), Attachment 8, page 6, states that laboratory scale model test measurements were used to develop multipliers to adjust the plant signal readings from the plant measurement location to arrive at an effective pressure at the dryer vertical face. Provide a detailed description of the scale model testing, including how the dryer loading was simulated and the results that justify the correlation of pressure values for different parts of the dryer. Confirm whether the test report has been submitted to the NRC and reviewed by the staff. If not, provide the scale model test report as part of the VYNPS EPU submittal.

## Response to RAI EMEB-B-5

The scale model test and results were briefly described in GE report GENE-0000-0018-3359-NP, "Technical Assessment, Quad Cities Unit 2, Steam Dryer Failure – Determination of Root Cause and Extent of Condition," dated August 2003 (ADAMS Accession No. ML032340379) which has been submitted to the NRC. A stand-alone report was not prepared for the scale model testing.

Following the lower cover plate failure at Quad Cities Unit 2 in 2002, a scale model test was developed in order to investigate the acoustic pressure loads acting on the outer surface of the steam dryer. Several phenomena have been suggested which could impose oscillating pressure loadings on the cover plate that failed. These include flow oscillations caused by turbulence, acoustic pressure oscillations, and disturbances caused by vortex shedding from various parts of the structure. Preliminary analyses showed that the vortex shedding frequency from typical structural members corresponded with low order acoustic frequencies. This observation suggested that the pressure oscillations associated with vortex shedding might excite a natural acoustic mode, creating large amplitude pressure oscillations on the dryer plate. Another suggestion involved periodic acoustic disturbances in the steam lines, which might cause an acoustic resonance in the dryer region. It is also possible that the frequency spectrum associated with turbulence (resembling white noise) also might have excited a natural acoustic vibration. The scale model test apparatus was designed to determine if vortex excited acoustic resonance is the likely root cause of pressure force oscillations that may have caused the plate failure.

The test apparatus models the steam dome region of the vessel and uses ambient air as the test medium. The scaled geometry consists of the outer surface of the dryer above the water level, the vessel head, the vessel wall, and the steam lines to the turbine inlet. Air is blown through the dryer up into the steam dome, where it then flows down the face of the outer hoods and out through the steam lines, mimicking the steam flow path in the plant. Pressure sensors (microphones) are mounted on the outer surface of the dryer at the cover plate and hood locations of interest. For the purposes of benchmarking the scale model results, additional pressure sensors are mounted at the locations where the in-plant sensors were located. The vessel diameter of the test apparatus is 14.5 inches, which translates to 1:13 to 1:17 scale, depending on the diameter of the plant being modeled. The test apparatus does not model the reactor internals or the dryer internals. A schematic of the test apparatus is shown in Figure EMEB-B-5-1. Typical sensor locations are shown in Figures EMEB-B-5-2 and 3.

Test Apparatus

The scale test apparatus can be separated into two primary components:

1. Test fixture
2. BWR mockup

The test fixture contains the components necessary to generate the required airflow and route the air to the BWR mockup. The mockup consists of the scaled steam dryer, RPV, and steam lines. The test fixture and mockup are described in more detail in the following sections.

Test Fixture: Figure EMEB-B-5-1 shows a schematic of the test fixture. The following components are identified on the drawing:

1. Blowers
2. Inlet Piping
3. Flow Meter
4. Muffler
5. BWR Mockup

The blower provides the system airflow. The air is routed through the inlet piping into the mockup. A venturi flow meter, and muffler have been mounted between the two skids. The venturi flow meter is used to measure the system airflow. The muffler is used to remove the noise introduced into the system by the test fixture components upstream of the mockup.

The second skid houses the BWR mockup components. Each of the BWR mockup components is scaled to represent the specific geometry of the plant being modeled.

BWR Mockup: The BWR mockup consists of three components:

1. RPV
2. Steam Dryer

3. Main Steam Lines

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The steam dryer mockup was fabricated using a fused deposition modeling rapid prototyping process. It was then nickel plated to prevent air from flowing through the mockup steam dryer surfaces.

The steam lines are fabricated from steel pipe. Each line contains an adjustable pipe section so that the effect of uncertainty in the plant line lengths may be investigated. The safety relief valve inlets will be fit with a length adjuster so that the effect of variations of the SRV inlet length can be evaluated. These components will be located in the correct scaled location along the pipe length.

Scaling Methodology

The scaling relationships used for the scale model are based on assuming a constant Strouhal number, preserving the Mach number between the small scale model and the full size system, and maintaining a consistent geometric scaling. With this approach, both the vortex shedding and acoustic frequencies are related by the length and velocity ratios. Frequency measurements in the scale model can be used to predict frequencies in the full size system. The frequency scaling used for this program is based on the relationship among frequency, wavelength and sound speed, as:

$$f\lambda = C$$

where:
$f$ = frequency
$\lambda$ = wavelength
$C$ = speed of sound

When the Mach number in the scale model matches the Mach number in the actual plant, the frequency scaling is

$$\frac{f_{Plant}}{f_{test}} = \left(\frac{D_{Test}}{D_{Plant}}\right)\left(\frac{C_{Plant}}{C_{Test}}\right)$$

where:
$D$ = characteristic length

The magnitude of the pressure oscillation is related to the magnitude of the velocity by:

$$P = \rho C U$$

where:
P = pressure
ρ = density
C = speed of sound
U = velocity

When the Mach number is preserved between the scale model and the actual plant, the pressure amplitude scaling is

$$\frac{P_{Plant}}{P_{Test}} = \left(\frac{\rho_{Plant}}{\rho_{Test}}\right)\left(\frac{C_{Plant}}{C_{Test}}\right)^2$$

The scale model testing is performed using airflow in a room temperature environment. As the air temperature changes, the speed of sound changes. Taking the basic equation:

$$C = \sqrt{k g_c R T}$$

where:
k = ratio of specific heats
$g_c$ = gravitational constant
R = gas constant
T = temperature

Using this to find the change in speed of sound with temperature, relative to a reference speed of sound,

$$\frac{dC}{CdT} = \frac{1}{2T}$$

For a temperature of about 530°R (70°F), this means the speed of sound changes by about 0.1% per degree Fahrenheit.

Methods & Assumptions

The test apparatus was designed using the following basic assumptions:

1. It is recognized that acoustic waves reflect from a water surface, just as from a solid wall. It is also recognized that the steam water interface in the vessel will not act as a completely reflective surface. It is expected that the bubbly interface will act as a partially absorptive surface; [[

]]

The water level between the dryer skirt and vessel wall forms the bottom of the cavity considered in the mockup. [[

]]

2. Acoustic waves originating in the steam/gas phase propagate into the water. However, if sound waves originate in water, they do not propagate into steam/gas. [[
]]

3. The model is only intended to be applicable for acoustic/flow phenomena. The model does not attempt to replicate any fluid structural interaction. It is assumed that the system acoustics and structural response are not coupled; in other words, the structural response does not significantly affect the fluid dynamics or acoustic properties of the system. Using this assumption, the scaled steam dryer and reactor assembly accurately replicates the dimensions of the acoustic cavity between the outer surface of the steam dryer and inside surface of the RPV; however, the wall thickness and materials are different.

4. It is assumed that the acoustic and the flow-instability phenomena (such as vortex shedding) are both well determined by scaling geometrically with the Mach number of the flow; therefore, the test apparatus was designed such that the Mach number in the test apparatus and the plant are equivalent. The scaling relationships discussed above assume a constant Strouhal number between the full scale plant and the scale mockup. Considering the scale used for the test apparatus, the Reynolds number in the scale model is not equivalent to that in the plant.

5. The Reynolds number (e.g. turbulence) may have some role in determining the flow rate at which acoustic standing waves are driven. The Reynolds number in the test apparatus is not equivalent to the Reynolds number in the plant; the scale model Reynolds number is approximately a factor of 500 less than that in the reactor. The scaling relationships derived for this program considered the effect of turbulence to be secondary.

6. The acoustic frequencies are determined by the overall dimensions such as the distance between the MSL nozzle and the top of the dryer hood. The overall geometry needs to be modeled because the shape of the open space in the steam plenum and main steam lines determines the overall acoustic mode shapes and hence acoustic frequencies. [[
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7. [[

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8. Steam with a moisture content of 0.1% by mass will have a density ratio of steam to water of about $4x10^{-5}$. This causes a very sparse distribution of very small water droplets in the steam. Considering the low moisture content, the speed of sound will be virtually unaffected by the moisture; therefore, it is acceptable to use air as the test fluid. There will be some slight attenuation of the amplitude over a long distance because of the droplets being oscillated by the acoustic waves.

9. [[

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10. [[

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11. The muffler in the scale model is located at a region that is not consistent with the steam water interface inside the stream dryer. Location of the instrumentation port prevents the muffler from being moved forward to a location consistent with the full scale plant. This difference between the scale model and the full scale plant can introduce a scale model specific acoustic mode that must be considered when reviewing the scale model data.

12. [[

]] The steam exiting the dryer banks flows upward into the steam dome and then is redirected to the outside of the vessel towards the main steam nozzles. The steam velocity increases significantly here as it passes over the edge of the dryer and enters the main steam lines.

Comparison Between In-Plant and Scale Model Test Results

A comparison was made between the in-plant and scale model test measured pressures between pairs of locations in order to justify the adequacy of the location multipliers based on the scale model test measurements. This comparison used the sensors located at the same elevation on the dryer skirt. A direct comparison of the location multipliers used in the generic load definition could not be made with the in-plant test measurements because the in-plant test had no pressure sensors on the vertical hood.

The location multipliers are ratios of the average pressure at one sensor location with the average pressure at another location. Sensor P1 was chosen as the reference location. P1 is located at an azimuth of 90° on the vessel skirt, which places it along the centerline of the outer vertical hood. Sensors P3 (azimuth 51°), P6 (azimuth 35°), and P7 (azimuth 10.5°) were each compared to P1 for each of the frequency ranges. Like the location multipliers, the comparisons were made in the form of a ratio P1/Px.

The pressure ratio comparisons are shown in Figures EMEB-B-5-4 through 6. As can be seen from the figures, the pressure ratios from the scale model test compare favorably with the pressure ratios from the in-plant test. The comparisons are quite good in the 0-55 Hz and 55-120 Hz frequency ranges, which is important because the fundamental structural frequency of the outer hood plates is in this range. These comparisons confirm the validity of the location multipliers developed based on the scale model test measurements.

The scale model test results are considered to be applicable to all dryer types and steamline configurations. Supplement 4, Attachment 7 page 10 Figure 2 shows the similarities in the measured in-plant test data for each of the three plants. The similarity in the in-plant test data suggests that the exact dryer type and steamline geometry are not significant factors in determining the fluctuating pressure loads on the dryer. It is expected that the scale model test would also show these similarities. [[

]] Therefore, the scale model test results are applicable to the VYNPS dryer analysis.

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## RAI EMEB-B-6

Supplement 4 (Reference 5), Attachment 8, page 18, Items 1 through 3 provide key assumptions used in developing the steam dryer fluctuating loads based on qualitative observation of measured plant data for several GE BWRs. The acoustic peak maximum amplitudes and frequencies of the acoustic peaks were assumed to be representative of all BWRs. It was also assumed that the maximum pressure amplitudes are related to the steam line flow velocity. Item 4 on this page states that the plant-specific scaling of the fluctuating loads is derived from the assumptions in Items 1, 2, and 3. Attachment 7, page 7, provides equations for determining the plant-specific load amplitude for each frequency zone. Provide information to benchmark the validity of these equations using the existing measured data.

## Response to RAI EMEB-B-6

The scaling exponents shown in the table in Attachment 7 page 18 will not predict the detailed response for each individual sensor used in developing the load definition. The acoustics that govern the measured response for each individual plant sensor are too complex to model in a practical generic load definition methodology. [[

]] The scaling exponents are reasonable for this purpose, as evidenced by benchmarking the load definition against the 2003 Quad Cities 2 experience. When the load definition is applied to Quad Cities at EPU conditions, the pressure loads are high enough to initiate fatigue cracking (Attachment 7 page 8). At the Quad Cities original licensed power conditions, the pressure loads are below the reverse-engineered pressures required to initiate fatigue cracking. This comparison substantiates the validity of the scaling exponents in the generic load definition.

## RAI EMEB-B-7

Supplement 4 (Reference 5), Attachment 8, page 7, states that the common BWR plant steam piping layout and the resulting similarities in the measured in-plant test data justify the application of the generic load definition to VYNPS. This appears to be in contradiction to the statement on page 148 of GE report GENE-0000-0018-3359-NP (see question 3) which states that the main steam lines and equalizing headers for different plants may have different as-built dimensions which could result in differences in pressure loading on the dryer. Please explain the apparent contradiction.

## Response to RAI EMEB-B-7

The two statements are not contradictory. Attachment 7 page 10 Figure 2 shows the similarities in the measured in-plant test data for each of the three plants. It is believed that the main steamline geometry plays an important part in determining the fluctuating pressure loads on the dryer. The similarities in the test data shown in Figure 2 are consistent with the common BWR steam piping layout. The as-built piping dimensions may play a role in determining the frequencies of. the specific peaks shown. For example, in the high frequency range (125-200 Hz), it is believed that the peaks are due to the acoustic excitation of branch lines on the main steamline (e.g., relief valve inlets). The length of the branch line determines the frequency of the peak; the sharpness of the inlet corners between the main line and the branch line determines how readily the resonance is excited and the resulting amplitude. Furthermore, the broad frequency zones used in the generic load definition are intended to bound the variations introduced by the individual plant as-built dimensions, thus making the generic load definition applicable to VYNPS.

## RAI EMEB-B-8

Supplement 4 (Reference 5), Attachment 7, page 7 states that scaling factors were determined for each frequency zone based on plant steam line velocity compared to the reference plant steam velocity. Provide an example to show how the scaling factors were calculated. It appears that the methodology does not address the type of steam dryers used and potential occurrence of the vortex shedding in the region between the dryer and the outlet nozzles. Past operational experience suggests steam dryer with square hoods have a higher frequency of failure than other types of dryers. Provide information to address the dryer geometry effects that cause the failure of the square type of dryers in the BWR plants. The QC2 dryer failures were, in part, due to the vortex shedding between the outlet nozzle and the outer hood. Address why the current evaluation at VYNPS does not include the performance of a computational fluid dynamic (CFD) analysis, which was previously used by GE to demonstrate the spacial pressure distribution and the reduction of pressure differential.

## Response to RAI EMEB-B-8

A sample calculation for the scaling factors is provided at the end of this response.

The range of steam dryer types is bounded by the in-plant test data used in developing the generic load definition. One foreign plant has a square hood dryer. The domestic plant and the other foreign plant have curved hood dryers. The slant hood dryer has characteristics in common with both of the other dryer types in that outer hood edge is square like the square hood dryer and the flow area between the outer hood and vessel wall is similar to that of a curved hood dryer. Attachment 7 Page 10 Figure 2 shows the similarities in the measured in-plant test data for each of the three plants. The similarity in the in-plant test data suggests that the dryer geometry is not a significant factor in determining the fluctuating pressure loads on the dryer.

The load due to vortex shedding from the top edge of the outer hood was postulated as a potential cause of the lower cover plate failure after three months of operation at EPU at Quad Cities Unit 2 in 2002. However, Dresden Unit 2 operated without failure for a full two year cycle. The pressure pulsations on the lower cover plate caused by the vortices will be about the same for the two plants because the dryer and vessel geometry in the outer hood region are virtually identical and the steam flow across the hood is about the same. The Dresden experience indicates that a load source or load combination other than vortex shedding alone may have been the cause of the 2002 lower cover plate failure at QC2. Based on the similarities in the measured data between the square hood and curved hood dryers shown in Attachment 7 Page 10 Figure 2, vortex shedding from the top edge of the outer hood does not appear to be a significant source of pressure loading on the dryer. If vortex shedding were a dominant mechanism, it is expected that there would be a significant difference between the in-plant test data for the square hood dryer and the curved hood dryers because of the difference in the geometry of the edge creating the vortices.

Even though the generic load definition methodology does not explicitly model potential vortex shedding from the top edge of the dryer, this effect is implicitly included in the load definition methodology. The in-plant sensors used in developing the load definition were located either on the dryer skirt or on the instrumentation mast above the dryer. Both of these regions are removed from the region between the outer hood face and the outlet nozzle; sensors in these regions would not pick up the pressures caused by potential vortex shedding in the outer hood region. As described in the response to EMEB-B RAI 5, the location multipliers used to adjust the plant signal readings to arrive at an effective pressure at the outer hood face were based on the scale model test measurements. The scale model measurements were taken at the in-plant sensor locations and on the vertical face of the outer hood. The sensors on the vertical face measure the pressures from all the load sources acting on the face, including acoustic, vortex shedding, and turbulence. Since the location multipliers are based on the ratio of the model pressures between the two locations, the effects of all the load sources are included in the multipliers. When the multipliers are applied to the in-plant measurements, the resulting effective pressure load on the outer hood face will include these load sources. Therefore, the effect of vortex shedding in the region between the outer hood and outlet nozzles is included in the generic load definition.

The key geometry feature in the square hood dryer type that led to the failures in the Quad Cities dryers are the 5" x 7" internal gussets that serve as attachment points for the internal diagonal braces. These internal gussets cause a very high local stress concentration in the outer hood plates. For the same pressure loading, the local stresses in the hood plates at the corners of these gussets are at least twice as high as the stresses in a square hood dryer without the gussets. Fatigue cracks initiated in the outer hood plates at these gusset corners in all four dryers at the Dresden and Quad Cities units during EPU operation. However, as evidenced by the successful operating histories of other BWR plants with square hood dryers at both original licensed thermal powers and EPU conditions, the presence of these internal gussets alone does not mean that all square hood dryers with internal braces will fail like those at Quad Cities. The pressure loads acting on the hood must also be high enough to initiate the cracks. The fluctuating pressure loads on the dryer are believed to be a strong function of the steamline flow velocity. The Dresden and Quad Cities steamline flow velocities at both original licensed power and EPU are the highest in the BWR fleet. The local stress concentration at the internal gussets combined with the high pressure loading at EPU led to the hood failures at Quad Cities. Therefore, it is likely that only the square hood dryers at Dresden and Quad Cities were susceptible to failure at EPU conditions. The steamline flow velocity at VYNPS at EPU conditions is about the same as the flow velocity at Quad Cities at original licensed thermal power. Therefore, the fluctuating pressure loads acting on the VYNPS dryer at EPU are expected to be considerably lower than the loads acting on the Dresden and Quad Cities dryers at EPU. However, in order to preclude the possibility a dryer hood failure under EPU conditions at VYNPS, Entergy has implemented the dryer modifications described in EMEB-B RAI 1 in which the internal gussets in the outer hoods were removed.

As documented in the response to EMEB-B RAI 1, a CFD analysis was performed as part of the VYNPS dryer evaluation using the CFD code described in the response to EMEB-B RAI 11. The primary concern in the dryer structural evaluations is determining the susceptibility to high cycle fatigue failure during normal operation. The CFD evaluations provide static loads that do not contribute to fatigue; rather, these loads are equivalent to the mean stress used in fatigue evaluations. The fluctuating pressure loads provide the alternating stresses that lead to fatigue. The fatigue stress acceptance criterion used in the VYNPS dryer fatigue evaluation was based on the fatigue limit curve that assumes the maximum allowable mean stress. Since the effect of the static loading was accounted for in the fatigue acceptance criterion, the static loads from the CFD evaluations did not need to be included in loads used in fatigue evaluation of the dryer.

<u>Sample Scaling Factor Calculation</u>

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NON-PROPRIETARY INFORMATION

**Table EMEB-B-8-1**
**Average Pressures, 0-55 Hz**

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NON-PROPRIETARY INFORMATION

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## RAI EMEB-B-9

Supplement 4 (Reference 5), Attachment 7, page 8, indicates that the generic load definition and scaling process used for the VYNPS plant-specific application compared well with the loading determined in the QC2 dryer failure root cause evaluation in 2003. In light of the subsequent failure at QC2 in March 2004, your argument does not provide reasonable assurance that the methodology is acceptable. In addition to the assumed acoustic loading, describe the potential flow induced vibration that may occur due to fluid elastic instability, vortex shedding, turbulence, two-phase flow impact, acoustic resonance and the possible fluid-structure interaction.

## Response to RAI EMEB-B-9

The cracking in the QC2 dryer repairs found during the March 2004 outage does not invalidate the generic load definition applied to the VYNPS dryer analysis. The cracking found in the 2003 QC2 dryer repairs resulted from local stress concentrations introduced by the as-installed repair configurations. Detailed evaluations of the as-installed repairs showed that when using the loading determined as part of the 2003 root cause analysis, the local stress concentrations were high enough to initiate the observed cracking and that the observed cracking was following the predicted stress fields. These observations serve to substantiate the load definitions used in the dryer analyses.

The pressure loads on the dryer are primarily acoustic in nature as evidenced by the sharp, well-defined peaks shown in Figure 2 of Attachment 7. At this time, the sources of the acoustic pressure loads are not well understood. In the low frequency range (below 100 Hz), vortex shedding from the top edge of the outer hood may provide the forcing function for the resonance peaks shown in Figure 2; interaction of the pressure pulses generated by the vortex shedding with the steamlines may affect the amplitude of the acoustic pressure loading on the steam dryer. In the high frequency range, the sharp acoustic resonances are believed to be caused by branch lines in the main steamlines. Vortex shedding caused by flow across the opening of the branch line excites a standing wave in the branch line cavity.

Pressure loading resulting from the turbulent flow across the dryer face is also acting on the dryer. The pressure loads due to turbulence are characterized by a broadband frequency spectrum with the amplitude decreasing as the frequency increases. The turbulence load amplitudes are small in comparison with the acoustic resonance loads. The minimum values of the pressure loads shown in Figure 7 show the turbulent pressure loading.

The generic load definition used in the VYNPS dryer analysis accounts for the loads due to vortex shedding, acoustic resonance, and turbulence. These loads are present in the in-plant test data used to develop the load definition. As described in the responses to EMEB-B RAI 5 and RAI 8, it is the intent of the scale model test to model these loads. The scale model test results for model sensor locations on the outer hood, skirt, and inner banks were used to develop location multipliers that adjust the in-plant pressure measurements taken at the dryer skirt and instrumentation mast to determine an equivalent pressure at the outer hood. The scale model measurements were taken at the in-plant sensor locations and on the vertical face of the outer hood. The sensors on the vertical face measure the pressures from all the load sources acting on the face, including acoustic, vortex shedding, and turbulence. Therefore, the location multipliers adjust the in-plant measurements to reflect the vortex shedding and turbulence pressure loads acting directly on the outer hood. Therefore, the generic load definition used for the VYNPS dryer analysis adequately accounts for these loads.

Fluid elastic instability and flutter can occur for strongly coupled fluid-structure systems when the feedback between the structural displacement and the resulting fluid flow field distortion are strongly correlated and grow without bound. Fluid elastic instability is usually associated with flow through heat exchanger tube bundles. Flutter is usually associated with airfoil structures. In both of these cases, a large structural displacement is required to interact with and distort the fluid flow field. Because the displacements in the dryer structure are small (on the order of tens of mils) when compared to the steam flow area between the dryer and the vessel, the dryer displacement will have a negligible effect on the flow field. Therefore, the dryer is not expected to be affected by phenomena such as fluid elastic instability or flutter[1].

Two-phase flow impact is only a concern for the main steamline break accident. During this event, the rapid vessel depressurization causes flashing of the water in the reactor vessel, resulting in a rapid level swell, which then impacts the dryer. As described in the response to EMEB-B RAI 11, the LAMB code is used to determine the pressure loading due to the two-phase flow impact on the dryer during the accident. The dryer is designed to maintain structural integrity for the two-phase level swell impact caused by a break of a main steamline outside containment.

The scale model test, generic load definition, and the finite element structural analysis assume that there is no fluid-structure interaction. Due to the small structural displacements, the structural motion of the steam dryer is not expected to be generating or contributing to the pressure load measurements shown in Figure 2. In general, the motion of the fluid and structure tend not to be in phase. The motion of the fluid would tend to inhibit the motion of the structure, thus dampening the structural response and reducing the overall stresses in the structure. Therefore, it is conservative to neglect the effects of fluid-structure interaction in the dryer analyses.

## RAI EMEB-B-10

The application dated September 10, 2003 (Reference 1), Attachment 4, Page 3-11, provides information regarding the structural evaluation for the steam dryer. The maximum estimated stresses for the normal operating condition due to flow induced vibration (FIV) are provided in Supplement 4 (Reference 5), Attachment 7, Section 8.3, at the critical dryer locations for the outer cover plates, hood vertical and top plates, hood end and partition plates, and hood bracing gussets. Provide the calculated stresses and cumulative usage factors (CUFs) at the dryer critical locations discussed above and also at the support brackets for the design basis loads such as dead weight, seismic safe shutdown earthquake (SSE) event and the main steam line pipe break at the EPU conditions.

[1] References: Au-Yang, M.K., 2001, *Flow-Induced Vibration of Power and Process Plant Components*, ASME Press, New York, NY; Blevins, R.D., 2001, *Flow-Induced Vibration, Second Edition*, Krieger Publishing Company, Malabar, FL.

## Response to RAI EMEB-B-10

The calculated stresses for ASME load combinations of the following components are contained in the response to EMEB-B RAI number 1.

| Component | Response to EMEB-B RAI 1 |
|---|---|
| Outer Cover Plate | Table 8 |
| Hood Vertical Plates | Tables 9, 10 |
| Hood Top Plates | Tables 11, 12 |
| Hood End Plates | Table 14 |

As stated in Section 1.0 to the response to EMEB-B RAI 1, the outer hood bracing brackets are removed in the steam dryer modification at VYNPS. Therefore, stresses for ASME load combinations are not calculated for these components. In addition, as stated in Note 2 below Table 2 in the response to EMEB-B RAI 1, the dryer partition plates are not a critical dryer location. Therefore the stresses for ASME load combinations are not explicitly calculated for these components. However the stress levels for the partition plates would be significantly less than that of the hood vertical plates since the inner dryer components, such as the partition plates experience much lower loading than the steam dryer outer components.

The maximum alternating stress due to the combination of upset pressure, OBE and turbine stop valve pressure is less than 38,000 psi, which occurs at the repaired front hood (See Tables 9 and 10 to the response to EMEB-B RAI 1). From Figure I-9.2.1 of the ASME Code, the allowable cycle is 120,000 cycles. The design OBE stress cycle is 50 and 360 cycles for turbine stop valve closures. Therefore, the CUF due to operating transient is less than 0.05. Other dryer components will have a CUF less than 0.05.

There are four support brackets to support the steam dryer. The dryer brackets are 304 stainless steel with a Sm of 16,675 psi at 575° F. The dryer brackets are full penetration welded to the reactor pressure vessel wall. From the load combinations equations for the stress analysis as previously stated in the response to EMEB-B RAI 1, the maximum calculated stresses on each support bracket at CPPU conditions is as follows:

| | | |
|---|---|---|
| Bearing stress | = 5,555 psi < Sy | Acceptable |
| Shear stress | = 2777 psi < 0.6 Sm | Acceptable |
| Bending stress | = 14,583 psi < 1.5Sm | Acceptable |

## RAI EMEB-B-11

The application dated September 10, 2003 (Reference 1), Attachment 6, Table 1-1, "Computer Code used for CPPU [Constant Pressure Power Uprate]," lists ISCOR, LAMB, and TRACG computer codes for computing the reactor internal pressure differences (RIPD) which were used for evaluation of reactor internal components. Specify which of these computer codes were used to calculate RIPD for VYNPS CPPU conditions. Discuss how these computer codes account for the effects of velocities, turbulence and vortex shedding in the regions between the steam dryer and the outlet nozzles while calculating the pressure differential across the dryer. Identify other computer codes that were used in the VYNPS plant-specific evaluation for calculating pressure variations on the reactor internals for CPPU conditions. Confirm whether these computer codes, methodology and models were reviewed and approved by the NRC staff especially for calculating the reactor internal pressure differences. If not, provide technical justification for applicability and acceptance of these computer codes.

## Response to RAI EMEB-B-11

The ISCOR, LAMB, and TRACG codes were used to calculate the RIPDs across the reactor internal components for VYNPS at CPPU conditions. The ISCOR code was used to calculate the normal and upset condition RIPDs for all the components shown in Table 3-4 of Attachment 6, with the exception of the steam dryer. With respect to the steam dryer, the ISCOR code was used to calculate the reactor heat balance conditions for normal operating conditions; the resulting steam flow rate from the core was then used to calculate the pressure drop through the dryer vane banks using an empirical correlation. The resulting dryer vane bank pressure drop is shown in Table 3-4. The normal condition RIPD results in Table 3-4 are used as the basis for the upset condition RIPDs shown in Table 3-5 of Attachment 6. The LAMB code was used to calculate the faulted condition RIPDs for the components shown in Table 3-6 of Reference 1, Attachment 6. The TRACG code was used to calculate the flow-induced loads on the core shroud and jet pumps during a recirculation suction line break Loss-of-Coolant Accident. NRC approval for the use of these codes for calculating RIPDs is documented in Attachment 6, Table 1-1. In addition, the VYNPS dryer analysis documented in Supplement 4, Attachment 7 also uses the computational fluid dynamics (CFD) code CFX-5.6 to calculate the steady-state pressure distribution across the steam dryer hoods. CFX is a commercial CFD software program; the application of CFX is based on standard industry practices.

The ISCOR code does not model the steam dryer. The ISCOR code is used to calculate the reactor heat balance conditions for normal operating conditions. The steam flow calculated by ISCOR is then used in the dryer vane bank pressure drop calculation, whether the calculation is performed using an empirical correlation methodology, as was done for the EPU submittal, or as input to the dryer CFD model used in the Attachment 7 dryer evaluation. Both the empirical correlation method and the CFD model calculate approximately the same pressure drop for the flow through the dryer vane banks. The CFD model also calculates the additional pressure drop across the outer hood panels caused by the fluid velocity and acceleration from the dome region to the vessel steam nozzle. As applied in the VYNPS dryer analysis, the CFD model does not account for the effects of turbulence and vortex shedding in the regions between the dryer and the outlet nozzles. The CFD model was used to determine the static pressure differentials and spatial pressure distribution on the dryer. Turbulence and vortex shedding are addressed in the response to EMEB-B RAI 9.

The LAMB code is used to calculate the pressure difference across the dryer for the main steam line break faulted condition for EPU dryer structural evaluations. The LAMB code modeling includes the velocity pressure drop associated with the flow across the outer hood. The LAMB code does not model vortex shedding or turbulence in the region between the steam dryer and the outlet nozzles. These loads are small compared to the RIPD loads induced by the two-phase level swell impacting the underside of the dryer. Also, the two-phase level swell in the annular gap between the dryer skirt and vessel wall will quickly reach the steamlines and disrupt the flow pattern in the region between the dryer and the outlet nozzles, which will disrupt the loads resulting from any vortex shedding or turbulence. The two-phase mixture entering the steamline will also disrupt any acoustic pressure loading that may be coming from the steamlines. Therefore, the LAMB model is adequate for calculating the RIPDs on the steam dryer for the main steam line break faulted condition.

## RAI EMEB-B-12

The application dated September 10, 2003 (Reference 1), Attachment 6, Section 3.4.1, states that the main steam (MS) and feedwater (FW) piping would have increased flow rates and flow velocities in order to accommodate the CPPU. As a result, the MS and FW piping would experience increased vibration levels approximately proportional to the square of the flow velocities. The ASME Code (NB-3622.3) requires that piping be designed and tested under startup or initial service conditions, for ensuring that vibration of piping systems is within acceptable levels. Based on the data provided in Attachment 6, Table 1-2, the vibration may increase as much as 60% of the vibration at the current rated power condition. In light of recent experience with regard to the failures of an electromatic relief valve, small piping failures in MS and FW lines, and FW probe failure during EPU operation in BWR plants, provide evaluations of piping vibration due to increased flow rates at the EPU conditions. In addition to reactor pressure vessel internals, the piping systems of interest include the MS and FW piping and their attached piping systems (e.g., MS drain lines, electro-hydraulic control lines, relief valve vent lines, thermowells, sample probes). Discuss your evaluations of potential adverse flow effects on reactor pressure vessel internals, and MS and FW systems and components, from EPU operation; the results of your evaluations; and modifications planned or completed to avoid adverse flow effects from EPU operation. Describe your plan and schedule of the vibration monitoring program with regard to the power ascension, monitoring methods (installing accelerometers, using hand-held devices), strategic locations of monitoring, and acceptance criteria. Confirm whether the vibration monitoring will be performed for both MS and FW lines and branch lines piping and components in accordance with the ASME OM Code.

## Response to RAI EMEB-B-12

The piping and piping component steady state vibration monitoring programs for the planned EPU operation have not been finalized. The piping vibration monitoring program addresses flow-induced vibration of selected piping systems that will experience increased flow during EPU operating conditions. The piping components program addresses flow-induced vibration for selected in-line components and components mounted to these piping systems. Plans for each program are described below along with a description of work accomplished to date.

The evaluation and results of the reactor vessel internals for the effects of flow induced vibration at EPU operation is contained in Attachment 6, Section 3.4.2 of the September 10, 2003 application. Other than the modifications implemented to the steam dryer (see response to RAI EMEB-B-1), no other VYNPS reactor vessel internals components require modification to mitigate adverse flow effects from EPU operation.

Piping Vibration Monitoring Program

The piping steady state vibration program for EPU operation follows the guidelines of ASME OM -S/G-2000 Code, Part 3. The program will assess the flow-induced steady state vibration levels of selected piping systems that will experience increased flow during EPU operating conditions. The program will include branch lines and cantilevered small bore lines which industry experience has shown are vulnerable to high-cycle fatigue failures. The affected piping systems are classified into one of the following three vibration monitoring groups and evaluated accordingly:

Vibration Monitoring Group No. 1:

This includes the main steam (MS) and feedwater (FW) piping located in the drywell which is inaccessible during plant operation. These pipes will be monitored for vibration levels utilizing remote accelerometers. The accelerometers are temporarily mounted to the piping and are hardwired to a remote, stand-alone digital acquisition system (DAS) that is located just outside the drywell. A total of 31 accelerometers were installed during the Spring 2004 refueling outage on the MS and FW piping in the drywell. The accelerometer locations incorporate GE recommendations. The locations correspond to the predicted high vibration response locations based on free vibration analyses of the MS and FW piping.

In addition, a total of seven accelerometers were mounted to two MS lines and one FW line in the heater bay. The accelerometers are temporarily mounted to the piping and are hardwired to a remote, stand-alone DAS that is located just outside the Heater Bay. Two MS accelerometers are mounted on the MS B and D line loops downstream of the turbine control valves. These accelerometers also measure the MS vibration response that is applied to the nearby low point MS line drain lines. MS line drain lines at another BWR experienced flow induced vibration failure at the socket-weld connection to the MS line. The MS line drain lines are being evaluated for the applied MS line vibration levels. Three FW accelerometers are mounted on one FW line located downstream of a flow element that could produce flow induced vibration.

The piping vibration stress acceptance criteria are based on the ASME OM Part 3 guidance. The design basis for the MS and FW piping is the ANSI B31.1 Power Piping Code. The stress criteria in ASME OM Part 3 is based on the ASME Code Section III. Thus, the acceptance criteria for the vibration monitoring of the MS and FW piping have been modified, such that the stress indices have been replaced with stress intensification factors, consistent with ANSI B31.1. Meeting the acceptance criteria demonstrates that the steady state flow-induced vibration stress levels of the

MS and FW piping remain below the appropriate endurance limit of the piping material as defined in ASME OM Part 3. GE recommended that branch lines be excluded from further evaluation if the measured main piping vibration levels at the branch locations are found to be small. If any levels are found to be significant, the measured main piping response and/or the measured response from accelerometers mounted on the branch line will be used to evaluate the affected branch line(s).

Piping vibration monitoring was part of the power ascension test program for the Spring 2004 refueling outage. Vibration data was taken at the 50%, 75%, 90%, 92%, 95%, 97% and 100% of original licensed thermal power (OLTP) levels. Frequency spectra were generated for each power level. All spectra showed low acceleration levels. The maximum root mean squared (RMS) acceleration recorded was 0.083 grms at the 100% power level. The maximum displacement recorded was less than 0.0025" at the 100% power level. The projected MS and FW piping accelerations at the 120% EPU power level are estimated to be less than 0.13 grms.

During the Spring 2004 outage, non-destructive examination (NDE) of the MS drain line socket weld connections to the MS lines found no evidence of fatigue related indications.

During EPU power ascension testing, vibration data will be taken at 105%, 110%, 115% and 120% OLTP levels. The measured vibration levels will be compared to the acceptance criteria.

<u>Vibration Monitoring Group No. 2:</u>

The FW regulator valves and attached FW piping located downstream of the reactor feed pumps experienced significant vibrations early in the plant's life. The vibrations were eliminated by modifications made to the valves. These components are located in the accessible feed pump room and therefore will be monitored with a hand-held vibration meter. These components will be monitored during EPU power ascension testing at the 105%, 110%, 115% and 120% OLTP levels. Baseline readings will be taken at 100% OLTP. If the measured EPU data indicates that vibration levels are increasing significantly, the affected components will be further evaluated. The acceptance criteria will also be based on the ASME OM Part 3 guidance and as with the acceptance criteria for Group No. 1, the stress indices will be consistent with ANSI B31.1.

<u>Vibration Monitoring Group No. 3:</u>

Visual monitoring will be employed during EPU power ascension testing to determine if significant vibration is occurring at the MS, FW and condensate piping located in the heater bay. Visual monitoring will be accomplished by walk downs and/or cameras. The monitoring will be performed and recorded at the 105%, 110%, 115% and 120% OLTP levels. Baseline monitoring will be performed prior to EPU power ascension at 100% OLTP. If visual observations indicate significant vibration, the piping will be monitored with an appropriate monitor and Vibration Monitoring Group No. 2 acceptance criteria will be used. Walkdowns performed during the Spring 2004 refueling outage revealed no indications of damaged piping (including attached small bore lines), pipe supports and attached components. No loose nuts or bolts on the pipe supports were observed.

Two cantilevered small bore FW piping lines containing large valves were modified to a weld design with increased fatigue resistance during the Spring 2004 refueling outage. These lines are located close to the high pressure FW heaters, which were replaced. In conjunction with the heater replacement, the lines were removed and re-attached with a 2-to-1 weld leg configuration socket welds per the guidelines of EPRI Report TR-113890, "Vibration Fatigue Tests of Socket Welds", dated December 1999.

As-built data and photos were taken during the outage to support evaluation of cantilevered small bore lines (2" diameter & less) with socket-welded connections. This information will be used to identify any configuration that should be visually monitored. Cantilevered lines that contain large valve(s) are candidates for monitoring. If significant vibration is observed, the appropriate monitoring and evaluation will be performed in accordance with the ASME OM guidance.

Piping Components Vibration Process

This program will identify and evaluate plant components and subcomponents (internal mechanisms) susceptible to flow induced vibration at power uprate operating conditions. The component types are: in-line (e.g., thermowells, etc.) components, directly mounted to piping that experience higher flow rates at EPU operating conditions (e.g., valves, pumps, instruments, etc.) and any instrument lines attached to the mounted component. These evaluations may require modifications to the component(s) and/or ongoing monitoring to minimize the potential for flow induced vibration failures.

The process steps to identify and evaluate susceptible components include:

- Reviewing EPU evaluation results for components identified as susceptible to vibration loading.
- Reviewing INPO and BWROG industry database of EPU-related vibration failures and issues.
- Developing a VYNPS specific database of vibration failures and issues. Review maintenance records and the condition reports that document equipment performance history. Interview VYNPS system engineers, operators and maintenance personnel to identify past or current vibration concerns.
- Categorizing the components identified above by failure mode (e.g. flow induced, high cycle fatigue, improper installation, instrument calibration, operational cause, aging, wear, material condition, etc.). Screen out component failures that are not flow induced vibration related.
- Screening out components not located on a piping system that will experience increased flow at EPU rated operating conditions.
- Evaluating identified susceptible components and subcomponents by analysis, testing, experience, or a combination of the three. Components such as thermowells will be analyzed for vortex shedding. Historical performance information and/or any available seismic testing of similar components provides evidence that the component will not experience vibration concerns during EPU operations. Any component located on vibration monitored piping can be

evaluated for the measured vibration levels at the selected CLTP levels. If the component is accessible, hand-held vibration levels may be taken at 100% CLTP power level. Readings will be used to project vibration levels at EPU and determine whether a component needs to be monitored during EPU power ascension testing. If the evaluation concludes that projected component vibration could result in an adverse impact, the preventative maintenance of the component will be evaluated for potential increase in frequency and/or enhancement. The component may also be identified as a candidate for periodic inspection.

- Selecting components for EPU power ascension testing. These components will be monitored for vibration levels at the 105%, 110%, 115% and 120% OLTP power levels. Baseline data will be taken at 100% OLTP power level. Corresponding system flow rates will be recorded simultaneously. Vibration monitoring equipment will be installed prior to EPU power ascension testing. Cameras may be utilized to observe and monitor vibration levels. As discussed above, hand-held readings may be employed. If required, remote accelerometers could be utilized and connected to the piping vibration data collection system. Measured data or observations would be compared to available baseline vibration data.
- Performing modifications if monitoring of the component is impractical and/or the evaluation demonstrates that the component will experience vibration concerns at EPU conditions. Modification may be required for the component itself and/or its associated system. Examples of potential modifications are the addition of pipe supports and the change-out of the component or subcomponent with a more suitable design.

During the Spring 2004 refueling outage, Entergy performed baseline walkdowns of selected plant areas to identify any pipe vibration issues resulting from operation at the 100% OLTP power level. As stated previously, no signs of damaged, distorted or loose connections for attached components and pipe supports were observed. Entergy engineers also looked for apparent configurations that may be vulnerable to flow induced piping vibration. In the drywell, valves located on the MS, FW, HPCI and RCIC piping were examined and no apparent vulnerabilities were identified. The actuators and solenoids for the MS safety relief valves are remotely mounted from the valve, thereby removing any flow induced vibration concerns. In the MS tunnel area, the MS and FW lines are heavily supported and no significant vibration of these lines can occur or be transmitted to attached components or the HPCI and RCIC branch lines. The free vibration analysis of the MS and FW piping substantiate this conclusion. In the heater bay, supply tubing to FW system air-operated valves (AOV's) was examined. Failure of this tubing occurred at another BWR during power uprate conditions when pipe vibration produced excessive relative movements between an AOV and the tubing anchor point. Similar configurations at VYNPS were found and the affected tubing was replaced with flexible tubing to ensure no flow induced vibration concerns will be present at EPU operating conditions.

## RAI EMEB-B-13

The application dated September 10, 2003 (Reference 1), Attachment 6, page 3-9, states that in Table 3-6, the allowable loads are compared to the applied loads for the CLTP and CPPU conditions for limiting shroud repair components. However, Table 3-6 (page 3-38) shows "VYNPS RIPDs for Faulted Conditions (psid)." Confirm that the correct reference should be Table 3-7 (page 3-39), "VYNPS Reactor Internal Components - Summary of Stresses."

## Response to RAI EMEB-B-13

The correct reference is Table 3-7 (page 3-39), "VYNPS Reactor Internal Components - Summary of Stresses."

## RAI EMEB-B-14

The application dated September 10, 2003 (Reference 1), Attachment 6, Section 3.5, states that the MS and FW piping were evaluated for the EPU conditions. Provide a summary of results of analysis for both the current rated and the EPU conditions in comparison to the code allowable limits. On page 3-17 of Appendix 6, you indicated that some MS piping supports will be modified due to the increase in MS flow rate for the EPU. Provide the schedule for implementing the modifications of these supports. Confirm whether the EPU analysis for the MS piping reflect the modified piping configuration.

## Response to RAI EMEB-B-14

A summary of results for the main steam piping that was affected by the CPPU is provided in Table EMEB-B-14-1 (for inside containment piping) and Table EMEB-B-14-2 (for outside containment piping). The data provided include stress levels for both the current and the CPPU conditions along with the allowable stress limits. It should be noted that since the main steam piping temperature did not change as a result of CPPU, no additional thermal expansion pipe stress evaluations were required.

**Table EMEB-B-14-1**

**Main Steam Piping – Inside Containment**

| Line Number | Loading Condition | Current Stress (psi) | CPPU Stress (psi) | Allowable Stress (psi) |
|---|---|---|---|---|
| Line A | DW + P + TSV + E | 13,883 | 14,154 | 18,000 |
| | DW + P + TSV + E' | 20,354 | 20,494 | 27,000 |
| | DW + P' + TSV + E' | 21,439 | 21,579 | 30,000 |
| Line B | DW + P + TSV + E | 12,917 | 13,191 | 18,000 |
| | DW + P + TSV + E' | 19,627 | 19,662 | 27,000 |
| | DW + P' + TSV + E' | 20,712 | 20,747 | 30,000 |

| | | | | |
|---|---|---|---|---|
| Line C | DW + P + TSV + E | 13,091 | 13,254 | 18,000 |
| | DW + P + TSV + E' | 19,719 | 19,805 | 27,000 |
| | DW + P' + TSV + E' | 20,804 | 20,890 | 30,000 |
| Line D | DW + P + TSV + E | 13,911 | 14,197 | 18,000 |
| | DW + P + TSV + E' | 20,369 | 20,518 | 27,000 |
| | DW + P' + TSV + E' | 21,454 | 21,603 | 30,000 |

Notes:
DW = deadweight stress
P = longitudinal pressure stress considering design pressure
P' = longitudinal pressure stress considering maximum pressure
TSV = turbine stop valve closure stress
E = design basis earthquake stress
E' = maximum hypothetical earthquake stress

**Table EMEB-B-14-2**

**Main Steam Piping – Outside Containment**

| Line Number | Loading Condition | Current Stress (psi) | CPPU Stress (psi) | Allowable Stress (psi) |
|---|---|---|---|---|
| Lines A, B, C & D | DW + P' + TSV + E | 15,548 | 16,432 | 18,000 |
| | DW + P' + TSV + E' | 17,413 | 18,042 | 30,000 |

Notes:
DW = deadweight stress
P' = longitudinal pressure stress considering maximum pressure
TSV = turbine stop valve closure stress
E = design basis earthquake stress
E' = maximum hypothetical earthquake stress

A summary of results for the feedwater piping that was affected by the CPPU is provided in Table EMEB-B-14-3. The data provided include stress levels for both the current and the CPPU conditions along with the allowable stress limits.

**Table EMEB-B-14-3**

**Feedwater Piping**

| System Boundary | Loading Condition | Current Stress (psi) | CPPU Stress (psi) | Allowable Stress (psi) |
|---|---|---|---|---|
| Feedwater Pump Discharge to Heaters E-2-1A&B | Thermal | 19,653 | 20,243 | 22,500 |

The feedwater piping downstream of heaters E-2-1A&B was analyzed using a conservative temperature of 400°F in the current design basis pipe stress calculations, which bounds the CPPU temperature of 393.5°F. Hence, no additional evaluations of this piping were required to reconcile the CPPU temperature of 393.5°F.

The results of the EPU MS piping analysis required modification to two pipe supports. Both modifications consisted of replacing the existing pipe clamp with a new pipe clamp. Both of these modifications were implemented during the April 2004 refueling outage (RFO 24). These pipe support modifications do not effect the piping configuration; therefore, there are no piping configuration changes to reflect in the EPU analysis.

## RAI EMEB-B-15

Regarding the application dated September 10, 2003 (Reference 1), Attachment 6, Section 3.5, provide a summary of the evaluation for the reactor recirculation piping and components for which the flow may increase to accommodate the increase in thermal power. Include recirculation pumps and valves and their supports, which may require a modification to support the EPU at VYNPS.

## Response to RAI EMEB-B-15

At rated core flow, the required recirculation pump flow will increase by 553 GPM (1.7 % of rated pump flow) for EPU conditions. The current design of the recirculation system will accommodate the slight increase in flow and continue to operate within its design capability. Consequently, the EPU conditions are within the original design capability of the system equipment including the pump, valves, piping and supports. Therefore, no modifications are required for the recirculation piping and components.

## RAI EMEB-B-16

The application dated September 10, 2003 (Reference 1), Attachment 6, Table 3-7 and Section 3.3.2, qualitatively evaluates the reactor internal components such as top guide, fuel channel, steam dryer, feedwater sparger, jet pump, core spray line and sparger, and incore housing and guide tube, for the EPU conditions. Provide a quantitative evaluation by comparing the key parameters and design transients, loads and load combinations that are used in the design basis analysis report for stresses and CUFs in each component, against the EPU conditions. Confirm whether and how the design basis parameters envelop those of the CPPU condition.

## Response to RAI EMEB-B-16

In general, the qualitative evaluation was performed due to one or more of the following reasons: (1) the load due to EPU is bounded by the existing design basis; (2) the change in load due to EPU is deemed insignificant; (3) the existing stress margin is substantially larger than the load increase; or (4) a comparison can be made to a similar plant /component. A specific assessment for the RPV internals is as follows:

NON-PROPRIETARY INFORMATION

**Table EMEB-B-16-1**

| Component | EPU RIPDs (psid) | | | CLTP RIPDs (psid) | | | EPU $P_m + P_b$ (psi) | CLTP $P_m + P_b$ (psi) | Remarks |
|---|---|---|---|---|---|---|---|---|---|
| | Normal | Upset | Faulted | Normal | Upset | Faulted | | | |
| Top Guide | 0.62 | 0.71 | 1.0 | 0.53 | 1.14 | 1.4 | EPU stresses remain the same or bounded by the CLTP stresses. | Longest Beam Stresses:<br>12,173 < 24,000 (N & U condition)<br>17,287 < 36,000 (Emergency condition)<br>40,307 < 48,000 (Faulted condition) | • Dead Weight, Seismic and fuel lift loads were considered in the stress calculations. These loads remain unchanged for EPU.<br>• [[ ]]<br>• High stress margin exists for the Normal, Upset, Emergency, and Faulted conditions. |

NON-PROPRIETARY INFORMATION

**Table EMEB-B-16-1**

| Component | EPU RIPDs (psid) | | | CLTP RIPDs (psid) | | | EPU $P_m + P_b$ (psi) | CLTP $P_m + P_b$ (psi) | Remarks |
|---|---|---|---|---|---|---|---|---|---|
| | Normal | Upset | Faulted | Normal | Upset | Faulted | | | |
| CRD Housing | N/A | | | N/A | | | N/A. Qual. Assessment performed based on CLTP stress margin. | 14,990 < 15,800 ($S_m$) (Upset cond.)<br>22,030 < 23, 700 ($1.5S_m$) (Faulted cond.) | • The design pressure, seismic and fuel lift loads remain the same as those for CLTP.<br>• [[ ]] |
| Control Rod Drive (CRD) | N/A | | | N/A | | | N/A. Qual. Assessment performed based on CLTP stress margin. | 20,790 < 26,060 | • The design pressure and seismic loads remain unaffected for EPU. |
| Orificed Fuel Support | 24.4 | 26.8 | 33.0 | 23.43 | 25.8 | 33.0 | 15,349 < 23,370 (Faulted condition stress conservatively compared with Emergency allowable) | | • [[ ]]<br>• The seismic and fuel lift loads remain unaffected for EPU.<br>• [[ ]] |

NON-PROPRIETARY INFORMATION

**Table EMEB-B-16-1**

| Component | EPU RIPDs (psid) | | | CLTP RIPDs (psid) | | | EPU $P_m + P_b$ (psi) | CLTP $P_m + P_b$ (psi) | Remarks |
|---|---|---|---|---|---|---|---|---|---|
| | Normal | Upset | Faulted | Normal | Upset | Faulted | | | |
| Fuel Channel | 13.32 | 16.22 | 17.0 | 12.24 | 15.14 | 16.8 | See Remark | | • [[ ]]<br>• Seismic loads remain unaffected. |
| Steam Dryer (Hood) | 0.45 | 0.59 | 8.6 | 0.35 | 0.53 | 6.8 | See Remark | 48,450 < 60,840 (Faulted cond.) | • Seismic loads remain unaffected for EPU.<br>• [[<br>• ]] |
| Jet Pump (Beam Bolt) | N/A | N/A | N/A | N/A | N/A | N/A | N/A. Qual. Assessment performed based on CLTP stress margin. | 14,600 < 22,800 lbs. (Bolt Preload) | • Recirc. Pump drive flow and seismic loads remain unaffected for EPU.<br>• [[ ]] |

NON-PROPRIETARY INFORMATION

**Table EMEB-B-16-1**

| Component | EPU RIPDs (psid) | | | CLTP RIPDs (psid) | | | EPU $P_m + P_b$ (psi) | CLTP $P_m + P_b$ (psi) | Remarks |
|---|---|---|---|---|---|---|---|---|---|
| | Normal | Upset | Faulted | Normal | Upset | Faulted | | | |
| Jet Pump (Riser pipe elbow to thermal sleeve) | N/A | N/A. | N/A | N/A | N/A | N/A | N/A. Qualitative. Assessment performed based on CLTP stress margin. | 9420 < 25,350 (N & U cond.)<br><br>9800 < 38025 (Emergency condition)<br><br>31,000 < 60,400 (faulted condition) | • Pump drive flow, seismic loads, etc. remain unaffected for EPU.<br>• [[ ]] |

NON-PROPRIETARY INFORMATION

**Table EMEB-B-16-1**

| Component | EPU RIPDs (psid) | | | CLTP RIPDs (psid) | | | EPU $P_m + P_b$ (psi) | CLTP $P_m + P_b$ (psi) | Remarks |
|---|---|---|---|---|---|---|---|---|---|
| | Normal | Upset | Faulted | Normal | Upset | Faulted | | | |
| Jet Pump Diffuser (N & U, Emergency) | N/A | N/A | N/A | N/A | N/A | N/A | 1297 < 24,000 (Upset and Emergency conditions)<br>40649 < 48,000 (Faulted condition) | N/A | • The specified EPU stresses are based on the seismic, acoustic, and flow induced loads, as applicable.<br>• [[<br>•<br>]] |
| Core Spray Line & Sparger | N/A | N/A | N/A | N/A | N/A | N/A | N/A. Qual. Assessment performed based on CLTP stress margin. | 14,476 < 24,000 (Upset limit) | • The system flow and seismic loads remain unaffected for EPU.<br>• [[<br>•<br>]] |

NON-PROPRIETARY INFORMATION

**Table EMEB-B-16-1**

| Component | EPU RIPDs (psid) | | | CLTP RIPDs (psid) | | | EPU $P_m + P_b$ (psi) | CLTP $P_m + P_b$ (psi) | Remarks |
|---|---|---|---|---|---|---|---|---|---|
| | Normal | Upset | Faulted | Normal | Upset | Faulted | | | |
| Feedwater Sparger | N/A | N/A | N/A | N/A | N/A | N/A | See Remark | | • [[ ]]<br>• All other loads such as seismic and feedwater flow either remain unchanged or the effect is insignificant. |
| In-core Housing and Guide Tube | | | | | | | N/A. Qualitative Assessment performed based on CLTP stress margin. | 15,290 < 23,700 | • Seismic loads and core flow in the lower plenum remain unaffected for EPU.<br>• [[ ]] |

## RAI EMEB-B-17

The application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.1.2.3, states that under CPPU conditions, the blowdown flow rate would increase slightly due to the increase in subcooling in the water initially in the circulation loops. This section does not address the change of annulus pressurization due to the increase in steam and feedwater flow for the EPU conditions. Discuss the change of the annulus pressurization due to MS and FW line breaks while the steam and feedwater flow rate will increase about 20% for the EPU operation. Confirm whether you considered the changes in annulus pressurization, jet impingement, pipe restraint loads or fuel lift loads in the analysis of the reactor vessel and internal components that are affected for normal, upset, emergency and faulted conditions as discussed on page 3-5.

## Response to RAI EMEB-B-17

The design basis event for the sacrificial shield wall annulus pressurization is the single-ended rupture of the 28-inch recirculation suction line. This rupture results in the maximum amount of flow into the annulus between the reactor vessel and the sacrificial shield wall. All other lines passing through the annulus are smaller diameter pipes (e.g. 18-inch main steam and 10-inch feedwater), thus the amount of flow from a single-ended rupture in any other pipe would be less than that from the single-ended rupture of the 28-inch recirculation suction line. The original design basis calculations only determined the annulus pressure for a single-ended break in the 28-inch recirculation suction line.

The 20% increase in feedwater and steam flow rates during normal operation at EPU conditions has no effect on annulus pressure. Break flow rate is principally determined by reactor pressure (which is unchanged by EPU), the size of the pipe (which is unchanged by EPU), and reactor fluid conditions (which is only slightly different for feedwater). Therefore, the increase in feedwater flow and steam flow during normal operation at EPU conditions has no effect on break flow rate assuming a break in those lines.

As stated in Section 3.3.2 of Attachment 6 to the September 10, 2003 power uprate application, the loads included in the reactor internals structural evaluation include Reactor Internals Pressure Differences (RIPDs), seismic loads, flow induced and acoustic loads due to Recirculation Line Break - Loss-of-Coolant Accident (RLB-LOCA), fuel lift loads, flow related, and thermal loads. The use of these loads for evaluation of the VYNPS reactor vessel internals for CPPU is consistent with the current design and licensing basis of the VYNPS reactor vessel internals.

## Plant Support Branch (IPSB)

### RAI IPSB-A-1

The application dated September 10, 2003 (Reference 1), Attachment 7, provides the justification for exception to large transient testing (LTT). Discuss why LTT is not considered necessary in light of recent industry experience relative to steam dryer failures. Include in your response: (a) how operation at EPU conditions may be likely to cause high-cycle fatigue in safety-related plant components (e.g., due to high steam line flow rates); (b) how lessons-learned from the April 16, 2003, inadvertent opening of a power operated relief valve at QC2, and its role in the second steam dryer failure, may be affected by plant operation at EPU conditions; (c) the possibility that performing LTT may identify undetected latent flaws in plant components and equipment normally subjected to pre-EPU conditions; and (d) how information contained in GE Service Information Letter (SIL) No. 644 and NRC Information Notice 2002-26, were considered in the licensee's decision not to perform LTT.

### Response to RAI IPSB-A-1

Entergy has reviewed NRC Information Notice 2002-26 and maintains ongoing involvement with industry issues described therein. Since the steam dryer failures and crack indications that occurred at Quad Cities and Dresden due to high cycle fatigue were identified, extensive methods and analyses have been developed by GE and others to further understand the impact of increased main steam line vibrations on, not only the dryer, but also other components that are impacted by main steam line vibrations. This information has been shared with the BWR Owner's Group as well as VY.

The primary lesson-learned from these investigations has been the impact of acoustic resonance and vortex shedding vibrations at high steam velocities. The loads imposed by these vibrations can, if present with sufficient amplitude, result in the high cycle fatigue failures that were experienced at Quad Cities. Entergy has analyzed the impact of flow induced vibration on the VYNPS steam dryer at CPPU conditions and installed a pre-emptive strengthening modification for predicted high stress components.

The VNPS dryer design uses the same basic configuration as the Quad Cities dryers (square hoods with diagonal internal braces). Therefore, an extensive structural analysis of the dryer was conducted to identify potential vulnerabilities and make a pre-emptive strengthening modification for operation at EPU conditions. The analyses considered not only the type of normal operating loads related to the Quad Cities failures, but also loads from anticipated transients (e.g., turbine stop valve closure and stuck open relief valve) and accidents (e.g., main steam line break outside containment). The modifications resulting from these analyses are described in detail in the response to EMEB-B RAI 1.

Large Transient Testing (LTT) places transients on plant equipment based on events such as turbine trips, which do not occur frequently. The purpose of LTT is to confirm the integrated plant response assumed in the associated safety analysis. Since the steam dryer is not credited in any safety analysis, LTT would not confirm response of this component. The testing also causes an undesirable impact on power generation and may not reveal any "latent flaws" in components. The dryer failures observed at Quad Cities were caused by high cycle fatigue due to the fluctuating loads experienced during normal operation. A significant amount of time was required to accumulate the fatigue damage that resulted in the failures: two to three months in the case of the lower cover plate failure at Quad Cities Unit 2 in 2002, and approximately one year for the hood failures at Quad Cities Units 1 and 2 in 2003. LTT would not impose the type of fluctuating loads on the dryer, and the duration, that lead to the observed fatigue failures. The April 2003 inadvertent opening of the power operated relief valve was only a secondary contributor to the hood failure at Quad Cities Unit 2. The pressure transient caused by the valve opening is believed to have provided the small additional load needed to rupture the remaining 30 mil ligament in the already degraded dryer. The outer hood at Quad Cities Unit 1 failed in November 2003 during normal operation with no significant transient loading. The Quad Cities Unit 1 hood failure demonstrates that, given sufficient time, the hood at Unit 2 would have also failed during normal operation without the additional loading from the valve opening. Similarly, the flow-induced vibration failures of components in the main steam and feedwater systems (relief valves, small piping, probes) were caused by high cycle fatigue during normal operation. The short transient loads associated with LTT would not identify undetected latent flaws in components subject to fatigue unless the component was already on the verge of failure. Therefore, LTT is not believed to provide any additional significant information with respect to long-term flow induced vibration and fatigue issues. The response to EMEB-B RAI 12 describes the evaluations, modifications, and vibration monitoring program that will be implemented to address the adverse flow effects resulting from EPU operation.

GE SIL 644 Supplement 1 recommends inspections and pre-emptive modifications to cover plates in BWR/3 type dryers prior to operating at power uprate conditions. The responses to EMCB-A RAI 2 and EMEB-B RAI 2 discuss results of the Spring 2004 VYNPS outage dryer inspection. SIL 644 Supplement 1 recommends monitoring certain plant parameters (steam moisture content, steamline flows, RPV level and pressure) that may indicate significant cracking in the dryer structure. The VYNPS dryer has incorporated the recommendations of the SIL in its evaluations as discussed above and, because of the improvements to the structure of the dryer assembly, LTT would not be expected to reveal any additional vulnerabilities beyond those identified by the dryer structural analysis and pre-EPU inspection.

## License Renewal and Environmental Impacts Branch (RLEP)

### RAI RLEP-C-1

Does Entergy have any protective measures to prevent aquatic species from entering the intake area on Vernon Pond?

### Response to RAI RLEP-C-1

As a result of more than a decade of prior studies that support current practices, Entergy does not have specific, protective measures to prevent aquatic species from entering the VYNPS' intake area on Vernon Pond.

VYNPS can and does entrain and impinge aquatic species. Entrainment of fish eggs and larvae was monitored for over a decade beginning in 1972. Entrainment was determined to be insignificant by VY's Environmental Advisory Committee (EAC) and was removed from the required National Pollution Discharge Elimination System (NPDES) Permit monitoring program. The EAC is comprised of representatives from the Vermont Department of Environmental Conservation, Vermont Department of Fish and Wildlife, New Hampshire Fish and Game Department, New Hampshire Department of Environmental Services, Massachusetts Office of Watershed Management, Massachusetts Division of Fisheries and Wildlife, and the Coordinator of the Connecticut River Anadromous Fish restoration program of the U.S. Fish and Wildlife Service.

Fish impingement has been monitored annually since 1972, and is considered low. There is a spring and fall period of monitoring. In both seasons, weekly and 24 hour samples are collected. All fish are identified, weighed, measured, and enumerated. These data are summarized and reported in VY's annual report entitled "Ecological Studies of the Connecticut River, Vernon, Vermont." The EAC has established impingement limits for both American shad and Atlantic salmon. VYNPS has never approached the impingement limits for these species.

There are no state or federally listed (endangered or threatened) species in the Connecticut River available to be entrained or impinged at the VYNPS site.

### RAI RLEP-C-2

What affect will the EPU have on the local tax base? Will the EPU result in increased tax revenues for Windham county, due to an increase in VYNPS value? Will the EPU lower the probability of early plant retirement? Please provide a short description of the benefits and disadvantages to the local community if the EPU was implemented.

### Response to RAI RLEP-C-2

VYNPS' public school taxes are assessed and collected by the State of Vermont under special statute. VYNPS is assessed at the state level and is exempted from the traditional, local property tax levy. The State Education Tax is based on a tax rate schedule applied to levels of generation over a three-year average. Additional generation of electricity from EPU will result in proportional tax increases.

Entergy's contribution to the remaining local tax base is governed through the year 2010 by a Tax Stabilization Contract that was entered into by the Town of Vernon, Vermont and the owners of VYNPS on June 7, 2000. The contract was properly assigned to Entergy as the new owner. The contract sets forth the Total Listed Value to be utilized for each year through 2010 for purposes of assessment of Municipal Services property tax. The contract specifies in Sections 1.01 – 1.03 that this Total Listed Value applies to all real and personal property owned on April 1, 2000, and all real and personal property thereafter acquired, which is used in connection with the generation of electrical power through the nuclear fission process.

Entergy does not remit tax revenues directly to Windham County, Vermont. There are indirect tax revenues as a result of state income taxes, sales taxes, hotel and meals taxes and property taxes paid by Entergy's employees and contractors who reside in the area while working at VYNPS. In addition, an EPU-related revenue sharing agreement between Entergy and the State of Vermont may, depending on the discretion of the State legislature, result in additional improvements to the local area.

The EPU will increase the economic viability of VYNPS and thus increase the likelihood of remaining operational at least through the end of its current license term, thus providing economic benefits to the southern Vermont area in the form of high-paying jobs, as well as the continued availability of reasonably-priced power under the existing Power Purchase Agreement between Entergy and the Vermont utilities.

The tax and other benefits to the local community are described above. In the past year, the Vermont Public Service Board concluded that the EPU will not unduly interfere with the orderly development of the region, will have minimal impact outside the immediate area of VYNPS, and is consistent with the relevant town and regional plans. No disadvantages to the local community have been identified as a result of the EPU.

## RAI RLEP-C-3

What is the estimated dose to members of the public located offsite due to the projected 1.2% increase in the volume of liquid radioactive effluents following the EPU?

## Response to RAI RLEP-C-3

As noted in Section 8.1 of the Power Uprate Safety Analysis Report (PUSAR) supporting the Vermont Yankee Nuclear Power Station (VYNPS) Extended Power Uprate (EPU), EPU is projected to increase the processed volume of liquid radwaste by 1.2% of the current total. However, this is an increased volume of liquid radwaste generated that requires processing, and not an increase in liquid radioactive effluents.

In addition, as noted in Section 8.6 of the PUSAR, there were no liquid effluents in the referenced five-year time period. The zero discharge operating philosophy currently in place at the VYNPS site will not be impacted by EPU, since the small percentage increase in liquid radwaste due to EPU conditions was evaluated and determined to be within the designed system total volume capacity. In the event that a liquid discharge may be necessary in the future, the prevailing operating philosophy will encourage operators to minimize any release and its consequences.

Because no liquid radwaste discharges are expected, and EPU results in a very small increase in liquid radwaste generated, it is unlikely that there will be any dose to members of the public from liquid radwaste generated as a result of EPU.

## RAI RLEP-C-4

Describe any known or observed threatened or endangered species on the VYNPS site. Specifically address the following species known to occur in Windham County: Bald Eagle, Indiana Bat, and Northeastern Bulrush. Have any surveys or studies been conducted on these species?

## Response to RAI RLEP-C-4

There are no known or observed threatened or endangered species on the VYNPS site.

Specifically, the Northeastern Bulrush was placed on the federal endangered species list throughout its entire range in 1991, and the Indiana Bat was placed on the federal endangered species list throughout its entire range in 1967. While both species may occur in Windham County, Vermont, no formal surveys have been conducted by Vermont Yankee or by the State of Vermont on the VYNPS site.

A pair of nesting bald eagles has built a nest on a Connecticut River island in New Hampshire, less than 0.5 miles downstream of the VYNPS site. This pair has nested in the vicinity for the past five years, and always on an island in the River, resulting in the nest residing in New Hampshire. There are no known nesting Bald Eagles in the state of Vermont. While the eagles are routinely observed flying over VYNPS, they feed largely on fish, and spend most of their time on the Connecticut River. This pair of eagles is closely monitored by the New Hampshire Audubon Society on behalf of the New Hampshire Fish and Game Department.

The Vermont Agency of Natural Resources continues to monitor a few specimens of the Giant Solomon's Seal, a rare plant, located on the VYNPS site.

## Plant Systems Branch (SPLB)

### RAI SPLB-A-1

General:

Implementation of the proposed VYNPS EPU requires increased volumetric flow rates, which result in higher flow velocities in the existing piping systems for the CPPU conditions. Please provide the calculated flow velocities that will result due to the proposed EPU conditions, and compare them to the design criteria and industry guidelines for systems such as main steam and associated systems, condensate and feedwater system, and other balance-of-plant (BOP) systems that are affected. Also, discuss in detail any dynamic loading and water hammer affects that the EPU will have on system functional and design capabilities.

### Response to RAI SPLB-A-1

The CPPU evaluation of velocities in BOP Systems determined that the existing design was acceptable. Flow velocities also affect pipe wall thinning due to flow accelerated corrosion (FAC). Other factors including piping material, flow conditions, and water chemistry can affect FAC wear rates. These factors are considered and will continue to be considered as part of the FAC program. FAC is discussed in PUSAR section 10.7:

> "VYNPS has evaluated CPPU system operating conditions for changes in FAC effects on plant piping and components. Implementation of CPPU primarily affects moisture content, temperature, oxygen, and flow velocity. The magnitude of predicted wear rates increase and vary throughout the BOP piping due to increased flows, temperatures, and the moisture removal capabilities of plant equipment. ... Based on experience at pre CPPU operating conditions and previous FAC modeling results, CPPU operating conditions will result in the need for additional FAC inspections.
>
> The increase in MS *(Main Steam)* and FW *(Feedwater)* flow rates at CPPU conditions do not significantly affect the potential for FAC in these systems. Increases in the low measured wear rates are expected to increase proportionately with flow. Operation under CPPU conditions will require additional focus for the FAC inspection program for the Main Steam Drains, Moisture Separator Drains, and the Turbine Cross Around System piping. The Extraction Steam System piping at VYNPS is constructed of FAC resistant material."

Note: No new systems are required to be added to FAC program, rather the "need for additional FAC inspections" (stated above) refers to potential changes in monitoring frequency/number of data points evaluated as part of the current FAC inspections, based on any changes in predicated wear rates or component life.

The following discussions summarize the calculated flow velocities and acceptance criteria for CPPU conditions. Note that in many cases velocities were calculated using a 122% power PEPSE heat balance model, which bounds the 120% CPPU condition.

Main Steam Velocity

Main Steam flow velocity values for original design, current, and expected uprate conditions are contained in Table SPLB-A-1-1. The steam velocity will increase approximately 29% from current. The uprate flow velocity remains within Stone & Webster / industry guidance for steam velocities.

The predicted uprate pressure, temperature, and velocity operating conditions in the Main Steam piping are acceptable and considered to be within the current design for proposed CPPU (including 122% analyzed conditions). Incorporation of uprate parameters and continued monitoring in accordance with the VYNPS FAC program will continue for the CPPU conditions.

**Table SPLB-A-1-1**
**RPV Steam Outlet/Main Steam Parameters**

| Parameter | Design, Rated | S&W Current PEPSE | Current, UFSAR Fig. 1.6-2 | GE CPPU, 120% | S&W 122% PEPSE | Is uprate condition bounded? – |
|---|---|---|---|---|---|---|
| Reactor Core Power Level, MWt | 1593 | 1593 | 1593 | 1912 | 1951 | Info only |
| Power Increase from Original, MWt | N/A | 0 | 0 | 319 | 358 | Info only |
| **From Reactor** | | | | | | |
| Velocity, Ft./s. | 174 | 173 | 173 | 213 | 218 | Yes - Note 1 |

**Turbine Stop Valve Inlet Piping (18") Conditions**

| Parameter | Design, Rated | S&W Current PEPSE | Current, UFSAR Fig. 1.6-2 | GE CPPU, 120% | S&W 122% PEPSE | GE CPPU 125% | Is uprate condition bounded? |
|---|---|---|---|---|---|---|---|
| Steam Flow Rate, lbm/hr | 6,423,000 | 6,430,654 | 6,431,532 | 7,900,000 | 8,068,494 | 8,295,000 | Yes |
| Velocity, Ft./s. | 185 | 181 | 185 | 228 | 233.6 | 239.4 | Yes - Note 1 |

Notes: 1. Stone & Webster / accepted industry recommended velocity is ~250 ft/sec. maximum. Uprate velocity is bounded by this recommendation.

NON-PROPRIETARY INFORMATION

Extraction Steam Velocity

Table SPLB-A-1-2 summarizes extraction steam velocities. The extraction steam piping at VYNPS is constructed of FAC resistant material. The extraction steam flow velocities listed below are based on the analyzed 122% heat balance conditions (1950.9 MWt) assuming a condenser pressure of 2.25 in Hg. The accepted / recommended maximum velocities for saturated steam in this pressure range are 1,000 feet per minute per inch diameter, with a maximum velocity of 15,000 fpm and a minimum velocity of 4,000 fpm. For low pressure extraction (below 15 psig) a velocity range of 12,000 to 18,000 fpm is recommended.

**Table SPLB-A-1-2**
**Summary of Extraction Line Velocities**

| | Extraction Line to Heater 1 | | Extraction Line to Heater 2 | | Extraction Line to Heater 3 | | Extraction Line to Heater 4 | | Extraction Line to Heater 5 | |
|---|---|---|---|---|---|---|---|---|---|---|
| | Current PEPSE | PEPSE 122% | Current PEPSE | PEPSE 122% | Current PEPSE | PEPSE 122% | Current PEPSE | PEPSE 122% | Current PEPSE | PEPSE 122% |
| Velocity, fpm /ft per sec | 8,326 /139 | 9,061 /151 | 10,226 /170 | 11,033 /184 | 9,823 /164 | 10,685 /178 | 10,613/177 | 10,695 /178 | 14,551 /243 | 19,152 /319 |
| Max. Recommended Velocity (fpm / ft per sec) | 12,000 / 200 | | 10,000 / 167 Note 1 | | 15,000 / 250 | | 18,000 / 300 | | 18,000/ 300 Note 1 | |

Note 1. The 122% velocities in extraction lines to FW heaters 2 and 5, exceed the recommended maximum by approximately 10% and 6% respectively. Note that the extraction line to FW heater 5 velocity at 120% power is 17,760 ft/min (296 ft/sec) and is less than maximum recommended velocity. Considering that all the extraction lines are low-alloy steel, and therefore, more corrosion resistant than carbon steel piping, this uprate velocity is considered acceptable, with appropriate FAC monitoring. Currently, all extraction lines are monitored by the FAC program. This piping has not experienced excessive corrosion/erosion rates and no significant changes are expected due to CPPU.

Condensate and Feedwater Velocity

Table SPLB-A-1-3 provides the calculated piping velocities in the Condensate and Feedwater system based on flows at the analyzed 122% power level and with Condenser pressure at 5" Hg. These conditions were used as the bounding parameter to determine the CFW system velocities because the CFW mass flows (and hence, piping velocities) are greater in this CPPU case when the flows are compared to the other CPPU cases.

Each segment from the Condenser Hotwell to the Reactor Vessel supply line was evaluated at CPPU conditions and compared to pre-CPPU operating conditions. The resulting CPPU piping velocities are presented below and compared to acceptable industry standards including S&W standards.

Typical acceptable velocities based on industry experience --

- Feedwater Pump Suction -- 600 feet / minute = 10 ft / sec
- Feedwater discharge -- 1,200 to 1,500 feet / minute = 20 to 25 ft/sec
- Condensate Pump Suction -- 3 feet per second at pump runout

NON-PROPRIETARY INFORMATION

**Table SPLB-A-1-3**
**Condensate/FW Branch Line Velocity Evaluation Results**

| Line Branch Description | Velocity at 100% Power ft/sec (current) | Velocity at 122% Power ft/sec (Note 1) | Velocity Criteria ft/sec |
|---|---|---|---|
| Hotwell to Condensate Pump Suction Header | 4 | 4.5 | 5 |
| Condensate Pump Suction Header | 6 | 7.8 | 5 |
| Condensate Suction Header to Cond Pumps | 3 | 4.1 | 3 |
| Condensate Pumps to Discharge Header | 6 | 7.8 | 10 |
| Condensate Pump Discharge Header | 10 | 13.0 | 10 |
| Cond Pump Discharge Header to SJAE | 12 | 15.5 | 10 |
| SJAE to Steam Seal Exhauster | 10 | 13.1 | 10 |
| Steam Seal Exhauster to Cond Demins | 10 | 13.1 | 10 |
| Cond Demins to LP Heaters 5A and 5B Header | 10 | 13.1 | 10 |
| LP Heater Header to Individual LP Heaters | 7 | 9.4 | 10 |
| FW Heater #5 to FWH #4 | 8 | 9.6 | 12 |
| FW Heater #4 to FWH #3 | 8 | 9.8 | 12 |
| LP Heaters to RFP Suction Header | 8 | 10.2 | 10 |
| Reactor Feedpump Discharge to Common Discharge Header (Note 2) | 15 | 13.0 | 25 |
| Reactor Feedpump Common Discharge Header | 7 | 8.5 | 25 |
| RFP Common Discharge Header to Individual HP FWH (Note 3) | 12 | 15.2 | 25 |
| FW Heater #2 to FWH #1 | 13 | 15.7 | 25 |
| Individual HP Heaters to FW 16x18 Reducer | 13 | 16.2 | 25 |
| From 16x18 Reducer to Reactor Vessel Supply Lines | 16 | 20.5 | 25 |
| Reactor Vessel Supply | 18 | 22.9 | 25 |

Note 1: As stated in Section 10.7 of PUSAR, feedwater flow rates at CPPU conditions do not significantly affect the potential for FAC in these systems. Increases in the low measured wear rates are expected to increase proportionately with flow. FAC wear rates are managed under the FAC Program.

Note 2: Based upon 2 pump operation at CLP and 3 pump operation at CPPU

Note 3: The maximum velocity in the FW system occurs at 18"x10" eccentric reducer at inlet and outlet of FWRVs. The velocity in these fittings is 34 ft/sec currently and increases to 43 ft/sec at CPPU. The fittings are currently monitored in the FAC program.

Feedwater Heater Drains Velocity

Table SPBL-A-1-4 provides the calculated piping velocities in the Feedwater Heater Drains System based on flows at the analyzed 122% power level and with Condenser pressure at 2.25" Hg and 1" Hg.

**Table SPLB-A-1-4**
**FW Heater Drain System Evaluation Results**

| Drain Flow Velocity from FW Heaters | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Equipment Conditions | Size Note 1 | Sch. Note 1 | Flow Area, sq.ft, Note 1 | No. of lines | Drain Flow, lbm/hr, total | Spec vol., cu. ft/lbm | Vol. Flow Rate, cu.ft/hr | Velocity, ft/sec (Note 1) |
| **FW Heaters 1A&B, HD-1A/B & HD-2A/B, Nor and Alt, Htr 1 to LCV** | | | | | | | | |
| 100%, Current | 6" | Std | 0.2006 | 2 | 374,388 | 0.0179 | 6,702 | 4.64 |
| 122%, 2.25" CPPU | | | | | 501,134 | 0.0181 | 9,071 | 6.28 |
| 122%, 1" CPPU | | | | | 501,756 | 0.0181 | 9,082 | 6.29 |
| **FW Heaters 2A&B, HD-3A/B & HD-4A/B, Nor and Alt, Htr 2 to LCV** | | | | | | | | |
| 100%, Current | 10" | Std | 0.5475 | 2 | 1,237,922 | 0.0179 | 22,159 | 5.62 |
| 122%, 2.25" CPPU | | | | | 1,539,413 | 0.0181 | 27,863 | 7.07 |
| 122%, 1" CPPU | | | | | 1,541,888 | 0.0181 | 27,908 | 7.08 |
| **FW Heaters 3A&B, HD-5A/B & HD-6A/B, Nor and Alt, Htr 3 to LCV** | | | | | | | | |
| 100%, Current | 14 | Std | 0.9575 | 2 | 1,630,730 | 0.0179 | 29,190 | 4.23 |
| 122%, 2.25" CPPU | | | | | 2,064,066 | 0.0181 | 37,360 | 5.42 |
| 122%, 1" CPPU | | | | | 2,074,241 | 0.0181 | 37,544 | 5.45 |
| **FW Heaters 4A&B, HD-7A/B & HD-8A/B, Nor and Alt, Htr 4 to LCV** | | | | | | | | |
| 100%, Current | 16 | Std | 1.2684 | 2 | 1,980,787 | 0.0179 | 35,456 | 3.88 |
| 122%, 2.25" CPPU | | | | | 2,491,524 | 0.0181 | 45,097 | 4.94 |
| 122%, 1" CPPU | | | | | 2,527,482 | 0.0181 | 45,747 | 5.01 |
| **FW Heaters 5A&B, HD-14 A&B, Nor and Alt, Htr 5 to LCV** | | | | | | | | |
| 100%, Current | 16" Branch | Std | 1.2684 | 2 | 2,421,084 | 0.0179 | 43,337 | 4.75 |
| 122%, 2.25" CPPU | | | | | 3,088,475 | 0.0181 | 55,901 | 6.12 |
| 122%, 1" CPPU | | | | | 3,239,069 | 0.0181 | 58,627 | 6.42 |
| 100%, Current | 20" Header | Std | 2.0142 | 2 | 2,421,084 | 0.0179 | 43,337 | 2.99 |
| 122%, 2.25" CPPU | | | | | 3,088,475 | 0.0181 | 55,901 | 3.85 |
| 122%, 1" CPPU | | | | | 3,239,069 | 0.0181 | 58,627 | 4.04 |

Notes:

1. Stone & Webster/Industry Heater Drain Line Design Criteria state that the maximum velocity for subcooled heater drain flow is 8 feet per second. All heater drain lines meet this criterion.

Moisture Separator Drain Velocity

Table SPLB-A-1-5 lists all the normal and emergency drain path flow velocities for current and uprate conditions. The velocity in these drain lines is evaluated based on flow of saturated water. The flow rates and specific volumes used to determine the velocities are from the current and 122% PEPSE heat balances. Pipe Section HD-13A-D exceeds this recommendation for current and CPPU conditions. This piping is inlet to the alternate drain path level control valve. The velocity is acceptable for this short piping section at the inlet to the control valve (LCV-103-23A-D).

### Table SPLB-A-1-5
### Moisture Separator Drain Line Velocity Evaluation Results

| Drain Flow Velocity from Moisture Separators | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **Equipment** Conditions | Size | Sch. | Flow Area, sq.ft, | No. of lines | Drain Flow, lbm/hr, total | Spec vol., cu. ft/lbm | Vol. Flow Rate, cu.ft/hr | Velocity, ft/sec (Note 1) |
| **Moisture Separators, HD-11A-D, 24" section, Capacitance adding pipe section** | | | | | | | | |
| 100%, Current | 24" | Std | 2.9483 | 4 | 701,532 | 0.0184 | 12,908 | 0.30 |
| 122%, 2.25" CPPU | | | | | 809,548 | 0.0184 | 14,896 | 0.35 |
| **Moisture Separators, HD-11A-D, 6" section** | | | | | | | | |
| 100%, Current | 6" | Std | 0.2006 | 4 | 701,532 | 0.0184 | 12,908 | 4.47 |
| 122%, 2.25" CPPU | | | | | 809,548 | 0.0184 | 14,896 | 5.16 |
| **Moisture Separators, HD-12A-D, 6" section** | | | | | | | | |
| 100%, Current | 6" | Std | 0.2006 | 4 | 701,532 | 0.0184 | 12,908 | 4.47 |
| 122%, 2.25" CPPU | | | | | 809,548 | 0.0184 | 14,896 | 5.16 |
| **Moisture Separators, HD-12A-D, 8" section, Drain Tank Outlet** | | | | | | | | |
| 100%, Current | 8" | Std | 0.3474 | 4 | 701,532 | 0.0184 | 12,908 | 2.58 |
| 122%, 2.25" CPPU | | | | | 809,548 | 0.0184 | 14,896 | 2.98 |
| **Moisture Separators, HD-13A-D, 4" section, LCV-23 Inlet Emergency Drain** | | | | | | | | |
| 100%, Current | 4" | Std | 0.0884 | 4 | 701,532 | 0.0184 | 12,908 | 10.14 |
| 122%, 2.25" CPPU | | | | | 809,548 | 0.0184 | 14,896 | 11.70 |
| **Moisture Separators, HD-13A-D, 6" section, Emergency Drain** | | | | | | | | |
| 100%, Current | 6" | Std | 0.2006 | 4 | 701,532 | 0.0184 | 12,908 | 4.47 |
| 122%, 2.25" CPPU | | | | | 809,548 | 0.0184 | 14,896 | 5.16 |
| Note 1: Stone & Webster/Industry Heater Drain Line Design Criteria state that the maximum velocity for saturated drain flow is 4 feet per second. This velocity is exceeded for the 4" and 6" pipe sections for current and uprate conditions. Significant increases in the low measured wear rates are not expected. However, FAC monitoring will note any changes and the wear rates will continue to be managed under the FAC Program. | | | | | | | | |

Cross-Around Piping Velocity

As shown in Table SPLB-A-1-6, the expected flow velocity is expected to have minimal, if any change. The mass flow rate increases by as much as 26% above the current flow, however, the increase in pressure will cause a decrease in the specific volume, which reduces the volumetric flow rate. The net affect of the increase in mass flow rate and decrease in specific volume is a minimal change in flow velocity. The Cross-Around flow velocities are approximately 7,500 fpm for the 36 inch pipe sections and 11,800 fpm for the 30 inch sections; all within the Stone & Webster guidance (15,000 fpm for saturated steam in this size piping and pressure range). Therefore, the uprate flow velocities are considered acceptable. Continued Flow Accelerated Corrosion (FAC) program monitoring will ensure that any changes to the erosion/corrosion rates are captured and managed.

**Table SPLB-A-1-6**
**Cross-Around Piping Velocity Evaluation Results**

| Cross-Around Piping (HP Turbine Exhaust) | | | | | | |
|---|---|---|---|---|---|---|
| **Parameter** | **Design, Rated – 1593 MWt** | **"Tuned" PEPSE – 1593 MWt** | **UFSAR Fig. 1.6-2- 1593 MWt** | **120%, Rated, 1912 MWt** | **122% PEPSE – 1950 MWt** | **Is uprate condition bounded?** |
| **36- inch Section From HP Turbine to Moisture Separator** | | | | | | |
| Velocity, ft/min | 7,597 | 7,098 | 7,526 | 7,554 | 7,573 | Yes - Note 1 |
| Velocity, ft/sec. | 126.6 | 118.3 | 125.4 | 125.9 | 126.2 | See ft/min. |
| Note: 1. The Stone & Webster/Industry criteria is 15,000 fpm for saturated steam in this size piping and pressure range | | | | | | |

| Cross-Around Piping (Moisture Separator Outlet) | | | | | | |
|---|---|---|---|---|---|---|
| **Parameter** | **Design, Rated – 1593 MWt** | **"Tuned" PEPSE – 1593 MWt** | **UFSAR Fig. 1.6-2- 1593 MWt** | **120%, Rated, 1912 MWt** | **122% PEPSE – 1950 MWt** | **Is uprate condition bounded?** |
| **30-inch Section From MS to LP Turbine** | | | | | | |
| **Velocity** | | | | | | |
| Velocity, ft/min | Note 2 | 11,494 | 11,546 | 11,590 | 11,564 | Yes - Note 1 |
| Velocity, ft/sec | Note 2 | 192 | 192 | 193 | 193 | See ft/min. |
| Note: 1. The Stone & Webster/Industry criteria is 15,000 fpm for saturated steam in this size piping and pressure range<br>Note: 2. Conditions not shown on design heat balance | | | | | | |

| Cross-Around Piping (LP Turbine Inlet) | | | | | | |
|---|---|---|---|---|---|---|
| **Parameter** | **Design, Rated – 1593 MWt** | **"Tuned" PEPSE – 1593 MWt** | **UFSAR Fig. 1.6-2- 1593 MWt** | **120%, Rated, 1912 MWt** | **122% PEPSE – 1950 MWt** | **Is uprate condition bounded?** |
| **LP Turbine Inlet** | | | | | | |
| Velocity, ft/min | 11,689 | 11,583 | 11,576 | 11,624 | 11,610 | Yes - Note 1 |
| Velocity, ft/sec | 195 | 193 | 193 | 194 | 193 | See ft/min. |
| Note: 1. The Stone & Webster/Industry criteria is 15,000 fpm for saturated steam in this size piping and pressure range | | | | | | |

An assessment of all affected piping was performed in order to reconcile changes in operating data resulting from the implementation of the CPPU. These evaluations included an assessment for the potential impact to flow induced fluid transient loading events (e.g., water hammer and/or steam hammer events). The flow induced fluid transient assessments performed considered specific piping system design inputs/attributes such as the magnitude of system flow rate increases resulting from CPPU, piping and pipe support configurations, presence of fast closing valves, etc.

The initial assessment performed on the main steam piping system determined that a more detailed evaluation was required to reconcile the higher system flow rate, and its impact on pipe stress levels and pipe support loads resulting from a turbine stop valve closure event. Therefore, detailed computer analyses were performed on both the inside and outside containment main steam piping system, to document the pipe stress and support acceptability of this system for the steam hammer loads associated with a turbine stop valve closure transient event. The results of these analyses determined that the main steam piping remains within acceptable allowable stress limits. Two main steam pipe clamps were replaced as described in RAI EMEB-B 14. The main steam piping and supports will continue to meet its design requirements under CPPU conditions.

The assessments performed on the balance of affected piping systems determined that these systems also remain acceptable, with respect to potential water hammer and/or steam hammer flow induced transient issues.

## RAI SPLB-A-2

Flood Protection:

The application dated September 10, 2003 (Reference 1), Attachment 6, Section10.1.2, states:

> "The flooding is dependent upon the maximum water levels in the hotwells..... FW system changes have been evaluated and the flooding rate from a FW line break is acceptable."

Supplement 4 (Reference 5), Attachment 6, MATRIX 5, Page 6, SE 2.5.1.1.1
VY NOTE, Flood Protection, states:

> "The limiting flooding events at VYNPS, however, are not controlled by fluid volumes in tanks and vessels, but results from open cycle systems such as Service Water, Fire Water, and Circulating Water System."

Please address the following:

a) Explain the difference between the above two statements. What are the limiting flooding events at VYNPS that could affect the performance of structures, systems, and components (SSCs) at the CPPU conditions? Please provide justification and/or details of the VYNPS evaluation that concludes that the SSCs important to safety will continue to be protected from flooding and will continue to meet the requirements of draft General Design Criteria (GDC) 2 following implementation of the proposed EPU.

b) Explain whether VYNPS performed calculations and/or analyses to evaluate the affects of fluid volumes in tanks and vessels on flooding. If so, are they based on the total volumes of the tanks and vessels or some lesser amount? If such analyses were not considered to be necessary, explain the basis for this conclusion.

## Response to RAI SPLB-A-2

From an internal flooding perspective, implementation of EPU has no adverse effect on the current licensing basis requirement of the plant to safely shutdown following a pipe break that results in internal flooding. The requirements of draft GDC-2 continue to be met.

a) PUSAR Section 10.1.2 deals with flooding from a high energy, closed cycle feedwater line break that is dependent on hotwell level, but which as discussed above is not the limiting flooding scenario. Review Standard RS-001 safety evaluation template section 2.5.1.1.1 deals with flooding from open cycle systems. In the case of VYNPS these open system flooding events involve the Connecticut River as the water source and are the limiting flooding scenarios.

   EPU does not require an increase in flow from the open cycle water systems such as service water, circulating water and fire water which have the Connecticut River as a supply source. As in the current licensing basis, open cycle system line breaks continue to be the limiting flooding events at VYNPS that could affect the performance of SSCs. EPU implementation has no effect on open cycle system break flow rate or total flow from these breaks. The amount of water in the river easily envelopes the amount of water in a full condensate storage tank or in the condenser hotwell.

b) VY's evaluations show that EPU also does not change the amount of water in any closed system. The levels in the condenser hotwell and condensate storage tank remain the same. Breaks in lines fed from these sources do not increase the total amount of flood water following EPU implementation. A line break that completely drains a full condensate storage tank continues to be bounded by an open cycle service water line break, as does a feedwater line break.

## RAI SPLB-A-3

Turbine-Generator and Internally Generated Missiles:

The application dated September 10, 2003 (Reference 1), Attachment 6, Section 7.1, states that the high-pressure turbine has been redesigned with new rotor, diaphragms, and buckets to increase its flow passing capability. Please address the following:

a) Explain the impact that these modifications will have on the existing turbine overspeed protection features and requirements, and how protection from turbine overspeed protection will continue to be assured.

b) Explain why no changes are required for the turbine overspeed trip set-point.

c) Explain why/how equipment important to safety will continue to be protected from the effects of turbine missiles

## Response to RAI SPLB-A-3

a) Prior to installing the high-pressure turbine conversion (during the April 2004 refueling outage) to support EPU, the low pressure (LP) turbine rotors at VYNPS had previously been converted to the monoblock design from the original built-up design that was installed with the original construction of the plant. This LP rotor conversion from built-up to the monoblock design effectively increased total rotor inertia values by almost 20% over the original rotors. This large increase in inertia slows the acceleration rate of the machine should a load rejection event occur. Consequently, the estimated peak speed following a full load rejection was reduced almost 1.2% from the original estimated peak speed (109.95% versus 108.77% with LP monoblock rotors). GE refers to this estimated peak speed as "normal overspeed" or NOS. For NOS it is assumed that all protective steam valves and control systems have responded as intended to minimize the resulting peak speed.

As stated in the VYNPS EPU license amendment request, only the high pressure (HP) turbine steam path is replaced. With the replacement of the HP turbine, the maximum power rating is now ~ 20% higher than before, with only a small increase in rotor inertia since the LP monoblock rotors were installed. The new NOS value at EPU conditions is, 109.60%, which is less than the original NOS when the unit was first installed. Consequently, there is no need to adjust the design setting of the mechanical trip, which remains at 110.5 - 111.5% of rated speed, as there is still sufficient margin between the NOS value and the minimum mechanical trip setting. This margin should normally be at least 0.5%, and presently it is 0.9%.

GE also calculates a second overspeed value, referred to as, "emergency overspeed", or EOS. This is the estimated peak speed following a full load rejection event when it is assumed that the first line-of-defense valves and speed control systems completely fail. The unit would rapidly accelerate to the mechanical trip speed range, which would activate the trip function and close the main and intermediate stop valves. The resulting peak speed is called the emergency overspeed value. The limit for EOS is 120% of rated. Considering the LP monoblock and HP steam path conversions, the present EOS value is 119.2% of rated speed, which is very close the original value and is also fully acceptable.

Consequently, the uprated overspeed characteristics are all within GE's experience and within all operating limits. All protective systems will function as before with no loss of overspeed protection.

b) No changes are required for the mechanical trip setting range as there is still sufficient margin between the uprated NOS value and the minimum mechanical trip setting, and the EOS value is still below the limit of 120% speed with the maximum trip setting assumed.

c) As stated previously in this RAI response and in Section 7.1 of Attachment 6 to the VYNPS EPU application dated September 10, 2003, the HP and LP turbine rotors at VYNPS are of the GE monoblock design (i.e., integral, non-shrunk-on wheels). This is the confirmation of the Constant Pressure Power Uprate Licensing Topical Report (NEDC 33004P-A, Revision 4) disposition that a separate turbine missile analysis is not required for CPPU if the turbine rotors are of the integral, non-shrunk on wheel type. Since the turbine peak overspeed at EPU conditions is less than that of the original construction

VYNPS units, and because of the use of monoblock rotors, equipment important to safety will continue to be protected from the effects of turbine missiles.

## RAI SPLB-A-4

Turbine Gland Sealing System:

With respect to Supplement 4 (Reference 5), Attachment 6, MATRIX 5, Page 8, SE 2.5.2.3 VY NOTE, Turbine Gland Sealing System, please provide the basis, with respect to safety considerations, for the VYNPS CPPU determination that the system is capable of performing its intended function without modification.

## Response to RAI SPLB-A-4

As stated in the VYNPS UFSAR Section 1.6.1.4.4 and Section 11.4, the turbine shaft gland seal system includes steam seal regulators, exhaust blowers, and a condenser for control of shaft leakage. This system discharges noncondensible gases from the gland seal system to the station main stack through an advanced off-gas system which provides holdup time for decay of radioactive gases. Radiation levels in the advanced off-gas system and the station main stack are monitored and appropriate operator actions are taken upon receipt of abnormal radiation levels. These actions or response times are not changed with CPPU. As stated in Section 11.4 of the VYNPS UFSAR, the steam seal system is a power generation support system and has no safety design basis. The turbine sealing system is designed to Class II seismic design requirements. CPPU does not affect the seismic loading of any plant equipment.

The acceptability of the VYNPS gland seal system for operation at CPPU conditions was determined by comparing the CPPU system parameters with the CLTP system parameters. The steam seal system regulator setting for both CLTP and CPPU conditions is 4 psi. This unchanged setting leads to unchanged pressures in the steam seal system supply header for CPPU conditions. Based on discussions with NRC staff it was understood that safety considerations do not apply to this evaluation.

Tables SPLB-A-4-1 through 3 contain results of the comparison that was performed during the CPPU evaluation.

**Table SPLB-A-4-1**
**Steam Seal Header Parameter Comparison**

| | CLTP Parameters | | CPPU Parameters | |
|---|---|---|---|---|
| | FLOW (lbm/hr) | TEMP (F) | FLOW (lbm/hr) | TEMP (F) |
| HP Turbine N1 gland | 13,300 | 225 | 7,267 (1) | 227 |
| HP Turbine N2 gland leakoff to Moisture Separator | 7,800 | 375 | 6,192 | 406 |
| HP Turbine N2 gland | 7,830 | 260 | 9,421 (2) | 268 |
| LP Turbine "A" N3 gland | 2,560 | 225 | 2,261 (3) | 227 |
| LP Turbine "A" N4 gland | 2,560 | 225 | 2,261 | 227 |
| LP Turbine "B" N5 gland | 2,560 | 225 | 2,261 | 227 |
| LP Turbine "B" N6 gland | 2,560 | 225 | 2,261 | 227 |

Note (1) The N1 steam seal header flow is lower for CPPU because the packing land area on the original HP rotor did not utilize the high/low tooth design. The HP rotor installed for CPPU conditions was designed with high/low tooth packing in the N1 packing area resulting in better sealing capabilities.

Note (2) Due to a higher 1st stage pressure as a result of CPPU, there was an increase in the steam flow from the N2 packing area. Calculations were performed for EPU on the steam seal header line to Packing #2 and it was confirmed by GE engineering that the 6-inch diameter pipe can accommodate the higher flow.

Note (3) The steam seal header flows in the LP sections are approximately the same as originally designed due to the fact the LP turbine backpressure does not change as a result of CPPU.

NON-PROPRIETARY INFORMATION

**Table SPLB-A-4-2**
**Steam Seal Exhaust Parameter Comparison**

| | CLTP Parameters | | | CPPU Parameters | | |
|---|---|---|---|---|---|---|
| | STEAM FLOW (lbm/hr) | AIR FLOW (lbm/hr) | TEMP (F) | STEAM FLOW (lbm/hr) | AIR FLOW (lbm/hr) | TEMP (F) |
| HP Turbine N1 gland | 610 | 110 | 170 | 334 | 120 | 201 |
| HP Turbine N2 gland | 450 | 100 | 240 | 446 | 112 | 204 |
| LP Turbine "A" N3 gland | 900 | 360 | 190 | 774 | 203 | 199 |
| LP Turbine "A" N4 gland | 900 | 360 | 190 | 774 | 203 | 199 |
| LP Turbine "B" N5 gland | 900 | 360 | 190 | 774 | 203 | 199 |
| LP Turbine "B" N6 gland | 900 | 360 | 190 | 774 | 203 | 199 |
| Steam Packing Exhauster (condenser inlet) | 4,660 | 1,650 | 195 | 3,875 | 1,442 | 199 |

**Table SPLB-A-4-3**
**Steam Seal System Parameter Comparison**

| Parameter | CLTP (lbm/hr) | CPPU (lbm/hr) |
|---|---|---|
| Startup Flow (normal clearance) | 14,000 | 10,565 |
| Startup Flow (double clearance) | 28,000 | 25,306 |

The total steam seal header flow for CPPU conditions decreases slightly from CLTP conditions; therefore, the steam seal regulator is adequate for CPPU conditions. The steam seal supply header pressures are unchanged for CPPU; therefore the steam seal header relief valves are adequate for CPPU conditions. The steam packing exhauster flow at CPPU conditions is within the capacity of the steam packing exhausters. The CPPU temperatures in the seal steam headers and seal steam exhaust lines is within the 750 °F temperature rating of schedule 40 carbon steel; therefore, the system piping is adequate. The small increase in temperature of the inlet flow to the steam packing exhauster condenser is within the capacity of the steam packing exhauster condenser.

## RAI SPLB-A-5

Main Steam Supply System (MSSS):

With respect to Supplement 4 (Reference 5), Attachment 6, MATRIX 5, Page 9, SE 2.5.4.1 VY NOTE, Main Steam Supply System, please explain how the MSSS will continue to meet draft GDC-40 and draft GDC-42 following EPU implementation.

## Response to RAI SPLB-A-5

Draft GDC 40 and 42 require the safety features of the Main Steam System to be protected against the environmental and dynamic effects and missiles that might result from plant equipment failures or a loss of coolant accident (LOCA).

The environmental qualification of safety-related components is addressed in Section 10.3 of the PUSAR.

As discussed in response to RAI SPLB-A-1, the main steam piping is adequately supported for the dynamic effects of potential fluid transient events, including LOCA. Steam hammer events such as turbine stop valve closure and relief valve discharge events were considered.

As stated in PUSAR Section 3.5, piping systems potentially impacted by CPPU were evaluated and no new postulated break locations were identified. In addition, since CPPU conditions do not result in an increase in pressure considered in high energy piping evaluations, there is no increased pipe whip or jet impingement loads on related HELB targets or pipe whip restraints.

As stated in PUSAR Section 3.5, the pipe stresses and pipe support loads of the main steam piping system due to CPPU were evaluated and were documented to be within design limits.

Piping systems impacted by CPPU were evaluated and no new postulated break locations were identified. Hence, there are no new missile concerns originating from piping that will result due to CPPU. An additional discussion of internally generated missiles is provided in the January 31, 2004 submittal, Attachment 6 of Supplement 4 to Proposed Technical Specification Change No. 263.

Therefore, the Main Steam System will continue to meet draft GDC-40 and draft GDC-42 following CPPU implementation.

## RAI SPLB-A-6

Turbine Bypass System:

According to the Updated Final Safety Analysis Report (UFSAR), Section 11.5.2, "Power Generation Design Bases," the main turbine bypass system shall have a capacity of 105% of the maximum expected turbine design flow. The application dated September 10, 2003 (Reference 1), Attachment 6, Section 7.3, states that the turbine bypass valves were initially rated for a total steam flow capacity of not less that than 105% of the original rated reactor steam flow (i.e., 7.06 Mlb/hr). Whereas, at CPPU conditions, rated reactor steam flow is 7.906 Mlb/hr, resulting in a bypass capacity that is only 89% of the CPPU rated steam flow. Although the licensee concludes that the bypass capacity at VYNPS remains adequate for normal operational flexibility at CPPU conditions, this appears to be a change in the plant design and licensing basis which has not been specifically recognized and addressed in the submittal. Please explain this apparent discrepancy.

## Response to RAI SPLB-A-6

The original license power design capacity, 105% original turbine design steam flow, of the main turbine bypass system at the Vermont Yankee Nuclear Power Station (VYNPS), is exceptionally large for a GE BWR. Most GE designed BWRs have a main turbine bypass capacity of between 25% and 40% of original turbine design steam flow. The reason for the large original turbine steam bypass capacity at VYNPS is that the plant was originally designed to withstand a turbine trip at rated main turbine steam flow without causing an automatic reactor scram. In order to meet this original design, the main turbine bypass capacity was designed for 105% original turbine design steam flow. Also the main turbine bypass piping and the main condenser were designed for this capacity. The ability of VYNPS to withstand a turbine trip without automatic plant shutdown was abandoned early after the original plant licensing and is not the current licensing basis of VYNPS. This can be seen by observation of Section 11.5.4.2 of the VYNPS UFSAR that states:

> *Upon loss of unit load, the main turbine speed and acceleration protection systems initiate fast control and intercept-stop valve closure. The difference between the valve position demand from the controlling pressure regulator (EPR or MPR) and the actual valve position creates an "error" signal which acts to open the bypass system to the main condenser. <u>With reactor rated thermal power above 30%, an automatic reactor scram will occur which prevents violation of the Minimum Critical Power Ratio (MCPR) limit.</u>* (emphasis added)

The CLTP design of VYNPS is such that an automatic reactor scram signal is generated upon loss of unit load, main turbine trip or main generator load rejection, above 30% of rated thermal power (RTP). Note that as stated in Attachment 6, Section 5.3.2 of the VYNPS EPU application dated September 10, 2003, the CPPU power level above which an automatic reactor scram occurs upon main turbine trip or generator load rejection is 25% CPPU RTP. In addition, operation of the main turbine bypass valves is conservatively not credited in the reactor transient analyses for main turbine trip and generator load rejection either at current rated thermal power (CRTP) or at CPPU RTP. Therefore, it is concluded that CPPU does not change the VYNPS current licensing basis with respect to the main turbine bypass capacity.

As stated in Section 7.11.3.4 of the VYNPS UFSAR:

> *The Turbine Bypass System is designed to control pressure (a) during reactor vessel heatup to rated pressure, (b) while the turbine is brought up to speed and synchronized, (c) during power operation when the reactor steam generation exceeds transient turbine steam requirements, and (d) when cooling down the reactor.*

The evaluation of the VYNPS turbine bypass system for CPPU concluded that the CPPU turbine bypass capacity of 89% CPPU rated steam flow was adequate to address all four of the criteria stated in Section 7.11.3.4 of the VYNPS UFSAR. Therefore, it is concluded that VYNPS EPU submittal did address the change in relative design capacity of the turbine bypass capacity (105% for CLTP and 89% for CPPU RTP). The VYNPS design process for implementation of CPPU adequately addresses the required documentation changes, UFSAR update, drawing updates, procedure changes, etc.

## Probabilistic Safety Assessment Branch (SPSB)

### RAI SPSB-C-1

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, discuss the risk implications of relying on containment accident pressure for emergency core cooling system (ECCS) pump net positive suction head (NPSH) by addressing the following:

a) Describe how the containment accident pressure credit impacts the probabilistic risk assessment (PRA) success criteria and accident sequence modeling. Identify which PRA accident sequences lead to core damage as a result of inadequate containment accident pressure.

b) How is inadequate containment accident pressure modeled in the PRA? What failure mechanisms (e.g., equipment failures, operator errors, etc.) have been included? How have their probabilities been estimated?

c) How much does inadequate containment accident pressure contribute to the overall core-damage frequency? Provide numerical results, including the Fussell-Vesely importance measures and the risk achievement worths (RAWs) for each basic event whose occurrence results in inadequate containment accident pressure.

d) What core-damage frequency would result if the PRA took no credit for containment accident pressure?

### Response to RAI SPSB-C-1

a) The more conservative initial conditions assumed in the design bases calculations are responsible for identification of the need to rely on containment accident pressure for emergency core cooling system (ECCS) pump net positive suction head (NPSH) as necessary for successful mitigation of design-bases accident sequences, in comparison with the best-estimate thermal-hydraulic calculations performed in support of the VYNPS Probabilistic Safety Analysis (PSA) model. The major differences in the containment model used in the MAAP thermal-hydraulic analysis performed to support PSA success criteria versus the design basis analysis were:

   1. use of best-estimate values of heat transfer coefficient for the RHR heat exchangers,
   2. crediting internal passive heat sinks within the containment, and use of nominal vs. minimum suppression pool water volume,
   3. use of nominal decay heat rate from ANSI standard 5.1-1979 (i.e., 2-sigma uncertainty was not applied),
   4. use of nominal 100% initial reactor power rather than the 102% calorimetric uncertainty value, and
   5. use of a more conservative feedwater pump post-event runout model.

   It was concluded that pumps taking suction from the suppression pool had adequate NPSH without requiring suppression pool overpressure. No changes were necessary to credit operator action for closure of the torus vent valve to ensure adequate NPSH for the ECCS pumps taking suction from the suppression pool.

b) The following is a discussion of the PRA accident sequences which lead to core damage as a result of inadequate containment accident pressure. The VYNPS PSA model does consider the impact of rapid containment pressure reduction on ECCS NPSH following emergency containment venting. In the case of large and medium LOCA event sequences, if suppression pool cooling (top event TC) fails and the main condenser is not recovered (top event RM), top event VT is used to evaluate use of the hard-piped torus vent to accomplish containment heat removal. A hardened vent path is provided for certain beyond DBA sequences involving loss of containment heat removal. Loss of containment heat removal can be postulated to result in pressurization of primary containment beyond the design pressure and can eventually lead to containment failure.

The hard piped vent is used to vent containment before containment failure to ensure the availability of ECCS equipment which relies on the suppression pool inventory for suction. The vent path is directly from the torus air space to the plant stack. The vent consists of 8" pipe which is connected to one of the torus/drywell vacuum breaker lines and discharges into the 12" standby gas treatment system exhaust line which, in turn, discharges to the plant stack. The hard piped vent is initiated via rupture disk. A normally closed motor operated valve (MOV) is installed in the vent line and is used to control or terminate venting. Top event VT considers only the pressure relief function of the vent. Reclosure of the vent to control containment pressure in order to preserve low pressure coolant injection (LPCI) and core spray pump NPSH is evaluated under Top event AI. Top event AI is used to evaluate the success of long-term injection after containment heat removal challenges are assessed. Operator action AINPSH, "Operator fails to control vent and LP fails due to loss of NPSH," is included to model the inability to maintain adequate containment accident pressure for LPCI/CS ECCS pumps taking suction from the suppression pool. The Human Error Probability (HEP) for this action was estimated to be 1.1E-03. The EPRI method was used to estimate this probability, based on three separate contributors: (1) non-recoverable mistakes associated with misdiagnosis, wrong detection, procedures, etc.; (2) non-response errors; and (3) manipulative errors.

c) The Fussell-Vesely importance measure and the risk achievement worth (RAW) for operator action AINPSH are 7.841E-6 and 1.007, respectively.

d) The incremental core-damage frequency resulting if the PRA took no credit for containment accident pressure control (i.e., AINPSH guaranteed failure) is estimated to be < 1E-10/reactor-year.

## RAI SPSB-C-2

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, what indications would be available to the operator during a loss-of-coolant accident (LOCA) which could indicate abnormal ECCS pump performance, especially cavitation due to inadequate NPSH? What actions would the operator take in response to indications of inadequate ECCS pump NPSH?

## Response to RAI SPSB-C-2

Vermont Yankee Nuclear Power Station (VYNPS) procedure ON 3164, "ECCS Suction Strainer Plugging" provides guidance to the plant operators. Per this procedure, indications available to the operator during a loss of coolant accident which could indicate abnormal ECCS pump performance, especially cavitation due to inadequate NPSH include:

1. Pump motor current indication erratic or decreasing;

2. Decreasing pump suction pressure (read locally) with steady state torus pressure/temperature/level conditions;

3. System flow rate erratic or less than expected for the backpressure to which the system is discharging;

4. Frequent adjustments of ECCS system discharge valve to maintain a constant flow rate at steady state backpressure/level conditions;

5. Audible indications of pump cavitation, such as increasing vibration/rough operation; and

6. Possible opening or cycling of minimum flow valves in response to flow decrease caused by suction strainer plugging.

In response to indications of inadequate ECCS pump NPSH, operators would consider taking the following actions as necessary:

1. Remove from service or throttle flow from those ECCS systems not needed to restore and maintain EOP parameters. Consider securing one of two running RHR pumps within a single loop;

2. If possible, re-align the suction of the core spray pump(s) to the CST. Limit total core spray pump flow from the CST to 8,000 gpm to maintain adequate NPSH;

3. If an ECCS pump is aligned to the CST, initiate replenishment of the CST; and/or

4. Consider aligning the service water system or fire protection system to the 'A' RHR loop per the appropriate appendix of VYNPS Operational Emergency Procedure OE 3107, "EOP/SAG Appendices."

## RAI SPSB-C-3

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, have reactor vessel isolation events been considered as possibly more limiting than long-term suppression pool heat up following a LOCA for ECCS pump available NPSH (i.e., reactor vessel isolation with high pressure coolant injection (HPCI) unavailable and automatic depressurization system (ADS) activated to proceed to safe shutdown)? When is suppression pool cooling initiated with respect to ADS actuation?

## Response to RAI SPSB-C-3

In addition to LOCA and ATWS, the application dated September 10, 2003, Attachment 6, Section 4.2.6 discussed SBO and Appendix R fire events. These isolation events have been considered and the assumptions related to suppression pool cooling and reactor depressurization via the safety/relief valves are discussed below. In each case, the peak suppression pool temperature was less than the LOCA peak temperature of 194.7 °F. At pool temperatures less than the LOCA peak, ECCS pumps will have more available NPSH, all other things being equal.

Appendix R

One RHR train (i.e., one RHR pump, one RHR heat exchanger, one RHR service water pump) is placed in suppression pool cooling mode at 10 minutes after event initiation. Reactor depressurization is initiated in one (1) hour. Peak suppression pool temperature was calculated to be 189.5 °F.

Station Blackout

One RHR train (i.e., one RHR pump, one RHR heat exchanger, one RHR service water pump) is placed in suppression pool cooling mode at 60 minutes after event initiation. Reactor depressurization is also assumed to be initiated at 60 minutes after event initiation. Peak suppression pool temperature was calculated to be 187.9 °F.

Based on the above results, the large break LOCA presented in the application dated September 10, 2003, Attachment 6, Section 4.2.6, represents the most limiting case for ECCS pump available NPSH.

## RAI SPSB-C-4

Licensee letter BVY 99-45 to the NRC dated March 31, 1999, discussed issues related to the suppression pool water temperature analysis for VYNPS. The letter states that the decay heat model has been found to be acceptably conservative when a 2-sigma uncertainty is applied. Provide clarification of how the 2-sigma uncertainty was applied with respect to the 2% thermal power uncertainty.

## Response to RAI SPSB-C-4

The shutdown core power used in the EPU long-term containment analyses includes fission energy following scram, fuel relaxation energy (corresponding to the sensible energy in fuel due to its elevated temperature) and decay heat based on the ANSI/ANS 5.1 – 1979 decay heat model with additional actinides and activation products per GE SIL 636, Revision 1. The 2-sigma uncertainty adder is applied to the decay heat component (i.e., the ANSI/ANS 5.1 decay heat model with additional actinides and activation products per SIL 636, Revision 1).

The shutdown power is input to the analysis as a normalized value against the analysis initial core thermal power. The initial analysis core thermal power is assumed to be 2% higher than the rated core thermal power, per Reg. Guide 1.49. The code calculates the core thermal power at a specific time by multiplying the normalized shutdown power by the initial core thermal power. Therefore, the 2% rated thermal power uncertainty is added to the rated core thermal power, and. the 2-sigma uncertainty is an adder to the decay heat.

## RAI SPSB-C-5

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, were the recommendations of SIL 636 Revision 1 (related to the determination of decay heat) used for the containment calculations and the ECCS pump NPSH calculations?

## Response to RAI SPSB-C-5

Yes. The recommendations of SIL 636 Revision 1, allowances for miscellaneous actinides and activation products, were incorporated in the determination of decay heat for the VYNPS EPU conditions. Decay heat was calculated based on the ANSI/ANS-5.1-1979 standard with SIL 636 Revision 1 additional actinides and activation products and the additional conservatism of 2-sigma uncertainty. This conservative decay heat was then used in the VYNPS EPU containment calculations. The results of the containment calculations were used in the VYNPS EPU ECCS pump NPSH calculations.

## RAI SPSB-C-6

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, this section states that the debris loading on the suction strainers and the methodology used to calculate available ECCS NPSH for CPPU are the same as the pre-CPPU conditions. Please verify that there have been no changes since your December 29, 1999 letter to the NRC documenting your completion of the actions requested by NRC Bulletin 96-03. If changes have been made to the debris loading calculations, please describe these changes.

## Response to RAI SPSB-C-6

There have been no changes affecting the December 29, 1999 letter to the NRC documenting the actions taken in response to NRC Bulletin 96-03.

No insulation has been added or removed from the drywell that would adversely affect the results of the debris loading calculations. Permanent lead blanket shielding has been added, but the shielding has been shown to be not susceptible to material failure following a LOCA; therefore, the additional shielding will not increase debris loading on the ECCS strainers.

No additional paint has been added to the suppression pool chamber or drywell that would adversely affect the design inputs used in the debris loading calculations. The methodology to calculate the debris loading and strainer pressure loss for ECCS NPSH for CPPU is the same as the pre-CPPU methodology.

The VYNPS specific sludge generation values have changed. The sludge generation rate was increased from 53 lbs/yr to 88 lbs/yr for a one-time extension of suppression pool cleaning. The actual sludge generation results are based on the torus cleaning that occurred during the April 2004 refueling outage (RFO-24) that indicated that 75 lbs of sludge had been generated in the six years since the torus was last cleaned and painted in RFO-20. The new sludge generation rate is approximately 12.5 lbs/yr and is less that the sludge generation rate used in debris loading calculations. The ECCS pump NPSH margin calculation has conservatively not been revised to reflect the reduced sludge generation rate of 12.5 lbs/yr.

## RAI SPSB-C-7

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, provide the value used for the residual heat removal (RHR) heat exchanger K value. Verify that no change was made in this value from that used in the current licensed thermal power (CLTP) licensing basis analysis. Please identify this CLTP analysis. Describe the testing done (type of test and frequency) to assure that this value remains bounding.

## Response to RAI SPSB-C-7

The CLTP analysis is incorporated in VYNPS UFSAR Section 14.6.3.3.2, "Torus Temperature Response." The RHR heat exchanger K value used for EPU torus temperature analysis is 179.375 BTU/sec-°F. This value is an approximately 1.9% increase from the previous CLTP value of 176 BTU/sec-°F and is based on the higher RHR HX heat transfer resulting from increased torus water temperature.

The RHR HXs were previously tested and the test results analyzed per the guidance of NRC Generic Letter 89-13. This testing showed that the heat exchangers' performance easily met their design heat removal requirements (much more than 1.9%) based on once per operating cycle cleaning. In accordance with the provisions of Generic Letter 89-13 the heat exchangers are cleaned once per operating cycle to maintain this level of performance and additional performance testing is not required.

## RAI SPSB-C-8

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, please supply figures of the pressure available and pressure required as a function of time for NPSH for anticipated transients without scram (ATWS), station blackout and Appendix R similar to Figure 4-6.

## Response to RAI SPSB-C-8

The figures requested are located in the calculations submitted in response to RAI SPSB-C-26 and are listed below.

SBO: see Figure 2, VYC-2314, Rev. 0

Appendix R: see Figure 4, VYC-2314, Rev. 0

ATWS: see Figure 4.3, VYC-0808, Rev. 6, CCN 04

## RAI SPSB-C-9

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, what flow rates are assumed for the ECCS pumps for the short-term and the long-term NPSH analyses? Page 4-10 discusses "expected" flow rates. How is it assured that the flow rate won't be less than the assumed values?

## Response to RAI SPSB-C-9

For conservatism, the values used in the NPSH analysis for LOCA are upper-bound values. The upper-bound values are based on the statistical uncertainty associated with the flow measurements performed during periodic surveillances. These values are provided in Table SPSB-C-9-1.

**Table SPSB-C-9-1**
**LOCA Single Failure Sensitivity Case**

| | | |
|---|---|---|
| One RHR pump per loop | 7,400 gpm | short term and long term |
| Two RHR pumps per loop | 14,200 gpm | short term |
| One core spray pump | 4,600 gpm | short term |
| One core spray pump | 3,500 gpm | long term |

The upper-bound values are conservative for NPSH since they maximize the head loss in the piping from the torus to the pump inlet and the required NPSH is higher at higher flow rates.

To provide conservatism in LOCA calculations (e.g., core heatup), lower-bound flow-rate values are used.

Expected values would be between the upper- and lower-bound values. In that sense, the use of the term "expected flow rates" on page 4-10 of the application dated September 10, 2003, Attachment 6, Section 4.2.6, was inappropriate.

## RAI SPSB-C-10

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, what, if any, containment accident pressure would be required if a more realistic calculation of drywell and wetwell response to a LOCA was performed rather than the design basis analysis? For example, nominal reactor power, decay heat without the 2-sigma, realistic pump flows, credit for the effect of suppression pool temperature on required NPSH, best estimate RHR heat exchanger performance, no single failures, normal suppression pool water level, etc. The response to this question may be based on existing sensitivity studies or engineering judgment. The staff is not requesting a calculation.

## Response to RAI SPSB-C-10

The response to RAI SPSB-C-1 included a discussion of best estimate suppression pool temperature calculations using the MAAP code. An additional sensitivity case has been performed with the GOTHIC code and a modified DBA LOCA model, assuming no single failure of an RHRHX.

**Table SPSB-C-10-1**
**LOCA Single Failure Sensitivity Case**

| With RHRHX Single Failure | | Without RHRHX Single Failure | |
|---|---|---|---|
| Peak Wetwell Temp (°F) | Time (sec) | Peak Wetwell Temp (°F) | Time (sec) |
| 195 | 25,890 | 169 | 4,000 |

Crediting the additional RHRHX enhances suppression pool cooling, and containment overpressure credit is not necessary for the DBA LOCA.

NON-PROPRIETARY INFORMATION

VY has also performed sensitivity studies with GOTHIC on the CLB analysis key input parameters.

**Table SPSB-C-10-2**
**LOCA Single Failure Sensitivity Case**

| Input Parameter | Change | | Suppression Pool Estimated ?T(°F) |
|---|---|---|---|
| Decay Heat | Cycle-Independent | Cycle-Dependent | -2.0 |
| Long Term Vessel Recovery with Minimum Suppression Pool Cooling | 2 LPCI and 2 CS | 1 CS | -8.0 |
| RHR Flow (as it affects RHRHX Performance) | 6,400 gpm | 7,000 gpm | -0.6 |
| RHRSW Flow (as it affects RHRHX Performance) | 2,700 gm | 4,000 gpm | -4.8 |

The combined effect of each incremental change would be less than the arithmetic sum of each individual change. For example, the combined effect of using a cycle-dependent decay heat plus 4,000 gpm RHRSW flow would less than -6.8 °F, but greater than -4.8 °F (i.e., between -6.8 °F and -4.8 °F). The combined effect of all of the above, which would be representative of expected performance with only one of two RHR heat exchanges available, and assuming the design basis service water temperature of 85 °F and initial pool temperature of 90 °F, would be greater than -8.0 °F and less than -15.4 °F (i.e., between -8.0 °F and -15.4 °F). Therefore, the peak suppression pool temperature would be between 187 °F and 180 °F.

At the upper end of the range (i.e., 187 °F), some small amount of credit for containment overpressure would still be required. At the lower end of the range (i.e., 180 °F), credit for containment overpressure would not be required.

While VY expects these results to be accurate, the sensitivity calculations were not performed to QA program requirements. The results demonstrate some of the large conservatisms in the accident analyses for EPU.

## RAI SPSB-C-11

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, list the conservatisms included in the calculation of available NPSH and containment accident pressure and the value of each conservatism in terms of suppression pool temperature or containment pressure.

## Response to RAI SPSB-C-11

Available NPSH (NPSHA) is determined by the following equation.

$$NPSHA = (p_{Torus} - p_v)(144)\,v + Z - H_f - H_d - H_s$$

where
$p_{Torus}$ = torus pressure, psia
$p_v$ = vapor pressure of the pumped fluid, psia
$v$ = specific volume of the pumped fluid, cu ft / lb
$Z$ = elevation head, torus to pump suction, ft
$H_s$ = suction strainer loss, ft
$H_d$ = strainer debris loss, ft
$H_f$ = friction loss in suction piping, ft

The principal parameters in the calculation of available NPSH for ECCS pumps are torus pressure, suppression pool water level, vapor pressure (directly related to suppression pool temperature), and head loss due to flow through the debris bed, the strainer, and the piping between the torus and the pump inlet.

Conservatism in the Torus Pressure Calculation for NPSH Evaluations

Torus pressure is calculated in a manner to produce the minimum value consistent with the maximum value for suppression pool temperature.

Conservative inputs are used in the SHEX analyses, which minimize the torus pressure to be used in NPSH evaluations. This includes the use of inputs used to specify the initial conditions and the inputs used to describe the modeling assumptions.

Initial Conditions

The initial conditions are specified to minimize the amount of non-condensible gas initially in the drywell and torus and therefore reduce the torus pressure response. These initial conditions are as follows:

1. Initial drywell temperature is at the maximum value (170 °F).

2. Initial torus airspace temperature is at the maximum suppression pool operating temperature (90 °F).

3. The initial drywell and torus relative humidity is at 100%.

4. The initial drywell and torus pressures are at the minimum values (1.7 psig and 0.0 psig, respectively).

Modeling Assumptions

The modeling assumptions are used to minimize the torus pressure during the event calculation. These include assumptions related to mixing of break flow and containment spray flow with the containment atmosphere, which is discussed below.

For the short-term analysis case (time < 600 seconds) it is assumed that LPCI flow going to the broken recirculation loop is injected directly into the drywell. It is further assumed that the LPCI flow rate into the broken recirculation loop is at the runout flow rates with all LPCI pumps available. [[

]]
This assumption minimizes the drywell pressure response, which in turn reduces the torus pressure.

For the long-term analysis (time > 600 seconds) it is assumed that only one RHR system (one RHR pump, one heat exchanger, one RHR service water pump) is available in the drywell and torus spray mode. [[

]]

Conservatism in the Suppression Pool Water Level Calculation for NPSH Evaluations

Suppression pool water level (Z) varies during a LOCA. The initial level is based on the Technical Specification minimum volume of 68,000 cu ft. Level initially increases due to the addition of reactor water and feedwater. The temperature increases, then decreases slightly as water is pumped back to the reactor by the ECCS pumps and as the pool cools. The NPSH calculation is based on a conservative single value of water level corresponding to a pool volume of 77,640 cu ft. At the time of peak torus temperature, the calculated pool volume is 79,470 cu ft. The difference in water level corresponding to this difference in elevation is approximately 0.25 ft.

Conservatism in the Vapor Pressure Calculation for NPSH Evaluations

The vapor pressure ($p_v$) is based on the calculated suppression pool temperature. Therefore, all of the conservatism in its value is due to the conservatism in the calculation of the temperature. Using ASME Steam Tables, it can be shown that a 1 °F reduction in suppression pool temperature in the region of interest corresponds to approximately a 0.5 ft increase in available NPSH due to the corresponding drop in vapor pressure.

The response to RAI SPSB-C-10 addresses the question of the degree of conservatism in the calculation of the pool temperature. Removal of individual conservatisms in the analysis inputs was estimated to decrease the calculated peak suppression pool temperature by 0.6 °F to 8.0 °F. Therefore, the corresponding effect on available NPSH would be to increase it by 0.3 ft to 4.0 ft.

Conservatism in the Head Loss Calculation for NPSH Evaluations

The head loss due to flow is based on the maximum flow rates shown in the response to RAI SPSB-C-9. The total loss is based on the head drop across the debris bed on the ECCS suction strainer, the strainer itself, and in the piping from the strainer to the pump inlet. Each of the terms for the RHR and Core Spray pumps at the time of peak torus temperature is listed in Table SPSB-C-11-1.

**Table SPSB-C-11-1**

| Pump | Flow Rate (gpm) | Head Loss Due to Flow (ft), Current Design Basis | | | |
|---|---|---|---|---|---|
| | | Piping | Strainer | Debris | Total |
| Core Spray | 3,500 | 3.06 | 0.29 | 0.21 | 3.56 |
| RHR | 7,400 | 2.61 | 0.33 | 0.33 | 3.27 |

Nominal values for core spray and RHR pumps are approximately 3,000 gpm and 7,000 gpm, respectively. Since the head loss in the piping and the clean strainer is proportional to velocity squared, the use of the nominal values for flow instead of the maximums would reduce the losses by approximately 0.9 ft for core spray and 0.3 ft for RHR.

The specified values for the head loss due to debris are based on the original design basis calculations for the ECCS suction strainers. The calculation was revised to evaluate the effects of an increase in peak suppression pool temperature, an increase in the sludge accumulation assumption, and the sensitivity of the head loss to changes in fiber loading. The revised calculation showed that the maximum head loss occurred at a reduced fiber volume, but the total head loss was still within the current design basis values shown in Table SPSB-C-11-1. The revised calculations gave a value for core spray that was 0.02 ft lower than the design basis, and a value for RHR that was 0.09 ft lower than the design basis.

Summary

The highest value of the above parameters in terms of impact on available NPSH is suppression pool temperature. The cumulative effects of each of the terms affecting suppression pool temperature cannot be determined by adding each of the individual terms. However, it can be said that the cumulative effect of all the terms would reduce the peak temperature by more than 8 °F and thus increase the available NPSH by more than 4 ft because of the corresponding decrease in vapor pressure.

Other terms can be added, thus the cumulative effect of level (0.25 ft), nominal flow rate (0.9 ft for RHR and 0.3 ft for core spray), and debris loading (0.09 ft for RHR and 0.02 for core spray) would result in an increase in available NPSH of approximately 1.2 ft for RHR and 0.6 ft for core spray.

The overall effect of all the terms is thus on the order of 5 ft in available NPSH, or 2.1 psia in terms of absolute pressure, not including the effect of the conservatisms in the calculation of torus pressure, which were not quantified.

## RAI SPSB-C-12

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, what values of required NPSH are assumed for the ECCS pump for which containment accident pressure is required? Is one value used for each pump or is there a range of values for each pump? Verify that no temperature corrections are made to the required NPSH values.

## Response to RAI SPSB-C-12

The required containment overpressure shown on Figure 4-6 of the application dated September 10, 2003, Attachment 6, is based on the required long-term NPSH applied from 600 seconds after the beginning of the event. One value was used for each pump for long-term operation. There are no temperature corrections applied to the required NPSH values.

**Table SPSB-C-12-1**

| Pumps | Flow Rate (gpm) | Required NPSH (long-term) (ft) |
|---|---|---|
| One RHR pump per loop | 7,400 | 31.7 |
| One core spray pump | 3,500 | 29.6 |

## RAI SPSB-C-13

The following questions pertain to the application dated September 10, 2003 (Reference 1), Attachment 6, Sections 10.5.3, 10.6, and 10.9. A normally open torus vent line must be closed to retain containment accident pressure for adequate available NPSH. The staff is not aware of this being necessary for other licensees with Mark I containments that have to rely on containment accident pressure for adequate available NPSH.

a) Describe the configuration of the line the valve is in. Provide a drawing or sketch.

b) Where does this line vent to?

c) What is the normal function of this line?

d) Is this valve a containment isolation valve?

e) What automatic closure/open signals does the valve receive?

f) What actions would the operator take if this valve does not close?

g) What is the motive power for this valve? Verify that this motive power will be available for the LOCA, Appendix R fire, ATWS, and Station Blackout scenarios.

h) What is the surveillance frequency for testing this valve, and what testing (stroke testing/leak testing) is required?

i) Is there another (redundant) valve in the line which can be closed if this valve does not close when required?

j) At what point in the accident sequence will the operator close this valve? Why is this time acceptable? What is the stroke time of this valve, and is this time accounted for in the determination of adequate available NPSH?

k) What indications will the control room operator rely on to verify that the valve is closed? Will this indication be available during a LOCA, Appendix R fire, ATWS, or Station Blackout event?

## Response to RAI SPSB-C-13

a) The torus 3-inch vent valve SB-16-19-6B, is an air-to-open, spring-to-close, fail-closed-on-loss-of-power valve that is located in the line that vents the torus to the standby gas treatment system (SGTS) to the plant stack. The flow path consists of the 3-inch torus vent SB-16-19-6B to SB-16-19-6 through the SGTS system (note: the SGTS fan is not normally running), and out to the plant stack. See attached sketch Figure SPSB-C-13-1.

b) The line vents the torus to the standby gas treatment system to the stack.

c) The normal function of this line is to maintain the torus at atmospheric pressure. The vent path ensures the drywell to torus differential pressure of 1.7 psid can be maintained without approaching the high drywell pressure scram and ECCS initiation setpoint of 2.5 psig.

Torus pressure of 0 psi is the input parameter to the Mark I containment analysis. The torus normal operating pressure of 0 psi permits operation with minimum drywell normal operating pressure of 1.7 psig greater than torus pressure, but less than the maximum normal drywell operating pressure of 2 psig.

d) The valve is a containment isolation valve and closes on a PCIS Group III isolation signal.

e) The valve will automatically close on a high drywell (2.5 psi) signal, low reactor water level (127 in.) signal, high reactor building ventilation radiation (14 mr/hr), or high refueling floor radiation (100 mr/hr).

f) The operator would re-position the control switch for SB-16-19-6B to the AUTO/CLOSE position to close the valve. If the valve remains open, the operator would close or verify closed SB-16-19-6 (normally open) and SB-16-19-7 (normally closed) to isolate this vent path. Note: Both of these valves also isolate on a PCIS Group III isolation signal.

g) Since the valve fails closed by spring pressure on a loss of air or power, the motive force is available for the LOCA, Appendix R fire, ATWS and station blackout scenarios.

h) SB-16-19-6B is stroke time (closed) tested quarterly and is Type C leak rate tested during each refueling outage.

i) If SB-16-19-6B fails to close, SB-16-19-6 (normally open) and SB-16-19-7 (normally closed) would isolate this vent path. Both these valves also isolate on a PCIS Group III isolation signal.

j) The valve will automatically close when a PCIS Group III isolation signal is received during a LOCA or ATWS. It will also close on loss of power during a station blackout event.

The event of concern is an Appendix R fire event with no isolation signal present. (NOTE: See also RAI response to RAI IROB-B 1 on page 66 of 120 of the VYNPS Submittal dated Jan. 31, 2004.) In this event, the operators will enter VYNPS procedure OP 3020, "Fire Emergency Response Procedure," and the appropriate Emergency Operating Procedures (EOPs).

The appropriate sections of OP 3020 will be revised to state that, if a scram has been initiated, the operator is to manually initiate PCIS Group II and Group III isolations. This task is accomplished by positioning the control switches for the respective PCIS Group valves to the closed position and verifying closed indication. (The valves may already be closed if an automatic isolation was received coincident with the scram due to reactor vessel low water level). The torus vent valve is a PCIS Group III isolation valve and its control switch will be taken to the closed position (even if the valve is already closed due to an automatic isolation) and will be verified closed via position light indication. All of the PCIS Group II and Group III valves' control switches and position indications are located in the control room. VYNPS procedures (EOPs) currently direct the operator to verify isolations following a reactor scram. If a required isolation does not occur, the operator is directed to initiate the isolation.

The above operator actions are straightforward and there is ample time for successful completion. The time available to take this action is ~ 40 minutes from the time of the reactor scram. Following the control room initial response to a reactor scram, there should be no competing functions that would distract the operator from taking these actions. Operators are trained in these actions (initiating and verifying PCIS containment Group isolations following a reactor scram) on the plant simulator.

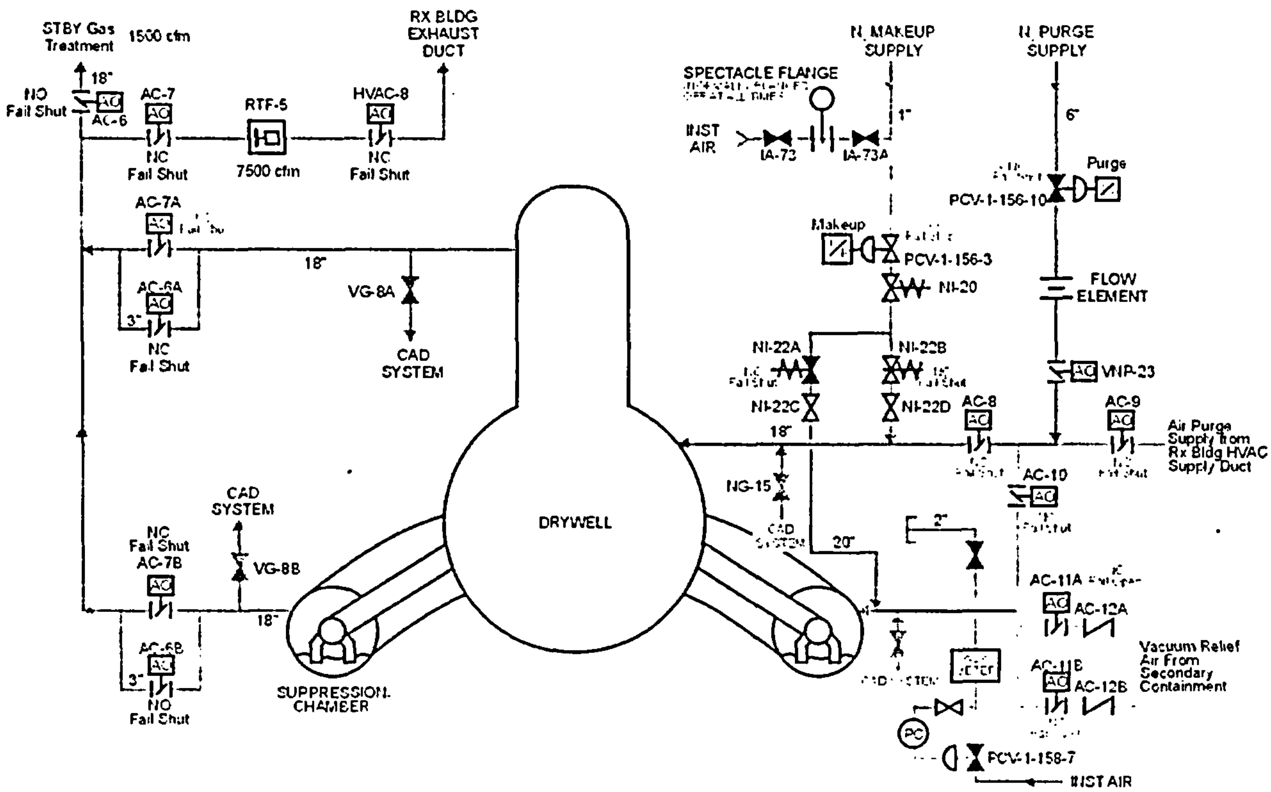
Indication of successful completion will be the closed position indication for the valve on the main control room front panel.

The inservice testing (IST) reference value for stroke time closed is 0.98 seconds. The IST acceptable range for this valve is 0.01 to 2.00 seconds. Since there are ~ 40 minutes available to take this action, the stroke time is acceptable. Closure of the valve within ~ 40 minutes will assure adequate containment overpressure due to subsequent torus heatup, if required, to assure adequate ECCS pump NPSH.

k) The position indication specifically for SB-16-19-6B is located on Control Room panel 9-4. Valve position indication is powered from instrument AC. This indication will be available during a LOCA, Appendix R fire, or ATWS event. For a station blackout event power is lost to instrument AC, causing SB-16-19-6B to close.

NON-PROPRIETARY INFORMATION

## Figure SPSB-C-13-1




*Containment Atmosphere Control System Simplified Drawing*

TRANSPARENCY 20 LOT-00-223 CR0392 REV051197

## RAI SPSB-C-14

The application dated September 10, 2003 (Reference 1), Attachment 6, Table 1-1, lists the computer codes used for CPPU. The NRC approved GOTHIC 5.0e for analyses performed for VYNPS Amendment 163. Table 1-1 states that GOTHIC 7.0 is being used for Appendix R fire protection analyses.

a) Please describe or provide a reference for the analyses and the assumptions used for these analyses.

b) The NRC issued a safety evaluation dated September 29, 2003 on the use of GOTHIC 7.0 for containment analyses for the Kewaunee Nuclear Power Plant, a Westinghouse-designed pressurized water reactor. Please verify that your use of GOTHIC 7.0 is consistent with the conditions specified in this safety evaluation.

c) Describe any other uses of GOTHIC 7.0 to support this power uprate.

d) Has an evaluation; in accordance with 10 CFR 50.59, been performed to evaluate whether NRC prior review and approval of the change from GOTHIC 5.e to GOTHIC 7.0 is necessary?

e) Has the guidance of Generic Letter (GL) 83-11 and GL 83-11 Supplement 1 been followed for the use of GOTHIC 7.0?

f) Describe any benchmarking done to support the use of GOTHIC 7.0.

## Response to RAI SPSB-C-14

Part (a):

Appendix R Suppression Pool Temperature Analysis Summary

For the current licensing basis (CLB) the GOTHIC code was used for calculating the transient suppression pool temperature for the limiting Appendix R scenarios. The GOTHIC model cases were updated with new decay heat and feedwater integrated mass versus feedwater enthalpy functions to reflect the EPU conditions. The peak suppression pool temperature for the EPU Appendix R limiting case is 189.5°F at 20,190 seconds versus 180.9°F at 20,400 seconds for the CLB.

Methods of Solution

Version 5.0e of the GOTHIC code was used in previous Appendix R analyses for VYNPS in support of license amendment 163. The latest version of GOTHIC is 7.0p2 and was selected for use in the EPU analysis. This specific version of the code has been installed and complies with the Entergy software QA procedure and the benchmark is documented in a safety related calculation.

Scenario Description, Assumptions, Inputs and Initial Conditions for Limiting Case

The limiting case Appendix R fire protection scenario for suppression pool temperature is a reactor building fire where normal shutdown cooling is unavailable. Alternate shutdown cooling is initiated from the main control room for cooldown. The scenario is modeled as follows:

1. Scram occurs at time zero.

2. The MSIVs are isolated at time zero (this is a conservative assumption for the suppression pool temperature calculation since the energy transferred to the condenser while the MSIVs are open will instead be transferred to the suppression pool when the MSIVs close).

3. Reactor vessel level is maintained by RCIC, HPCI or feedwater. For this analysis, because it results in the highest suppression pool temperature, the assumption is made that reactor vessel level is controlled by feedwater injection.

4. Suppression pool cooling is manually initiated 10 minutes after the scram occurs.

5. The feedwater system is aligned to bypass the feedwater heaters at 30 minutes into the transient in accordance with plant operating procedures. With the feedwater heaters bypassed, the temperature of the injected water will conservatively be 140°F.

6. At one-hour, an orderly reactor cooldown occurs at a rate of 100°F per hour using the SRVs. The higher cooldown rate is conservative because it maximizes suppression pool temperature.

7. Reactor water level is controlled with feedwater until all the feedwater is injected. When the pressure permissive is reached, core spray starts to inject into the vessel. After level is recovered to within the normal range, the core spray system is used to maintain RPV level, with vessel pressure being controlled by SRV cycling between 50 and 100 psig. The suppression pool is cooled continuously by the RHR system. The reactor vessel is maintained in this configuration for one hour. At this point the control room operators proceed with alternate shutdown cooling.

8. During alternate shutdown cooling, the operators increase the flow from the core spray pump to flood the vessel and the steam lines. The pressure in the vessel is maintained between 100 - 230 psig by plant operating procedures. The vessel is flooded with water spilling through an open SRV. The transient is run until the temperature in the suppression pool shows a turnaround (suppression pool temperature trends downward).

The key parameters and input assumptions are presented below:

1. The plant is assumed to operate at CPPU rated power, and an additional 2% calorimetric uncertainty is used in the analysis. This results in an assumed power level of 1950 MWt.

2. The CLB GOTHIC reactor vessel model assumes a full core of GE-9 fuel bundles. It was demonstrated that the stored energy in the reactor vessel for a full core of GE-9 fuel bundles bounds a full core of GE-14 fuel bundles at EPU conditions.

3. The ANSI/ANS 5.1-1979 decay heat with a 2-sigma uncertainty is used in the analysis.

4. A maximum allowed RHR service water (RHRSW) temperature of 85 + 0.1°F was used. The additional 0.1°F provides margin for RHRSW pump heat addition.

5. The RHR heat exchanger (RHRHX) performance was modeled conservatively (5% plugging).

6. The following tables provide the initial conditions used in the calculation. Table SPSB-C-14-1 presents the reactor vessel and core initial conditions; Table SPSB-C-14-2 presents the ECCS initial conditions and parameters; Table SPSB-C-14-3 presents the primary containment initial conditions; and Table SPSB-C-14-4 summarizes the timing of the limiting events.

**Table SPSB-C-14-1**
**Vessel and Core Initial Conditions**

| Parameter | CPPU Nominal Value | CPPU Analysis Value | Comments |
|---|---|---|---|
| Power | 1912 MWt | 1950 MWt | EPU plus 2% uncertainty |
| Decay Heat | ANS 5.1 | ANS 5.1 +2-sigma | |
| Vessel Pressure | 1020 psia | 1045 psia | The use of 1045 psia for the vessel pressure for the suppression pool temperature Appendix R analysis is conservative. |
| Vessel Level | 162 inches | 172 inches | Analysis value conservatively accounts for 3 inches above normal water level (uncertainty and operational fluctuations) and 7 inches for dimensional uncertainties. |
| MSIV closure time | 3-5 seconds | 3 seconds | The Technical Specification minimum value closure retains more energy in the vessel |
| Core Flow Rate | 48.0E6 lb/hr | 51.36E6 lb/hr | Increased core flow (ICF) conditions lead to higher vessel stored energy. |
| Initial Feedwater Flowrate | 7.88E6 lb/hr | 8.076E6 lb/hr | Feedwater flow consistent with power level (EPU) |
| Initial feedwater temperature | 393.5-393.6 °F | 393.9°F | Consistent with 20% power uprate +2% calorimetric uncertainty. |
| Hotwell temperature before heater bypass. | 110°F | 131°F | CLB nominal condensate temperature is 107.1°F at the outlet of the steam packing exhauster. This can be higher during summer months or lower during winter months corresponding to hotter or colder water in the circulating water system. The CLB summer hotwell peak temperature is 131°F; however, all these data are at the present power. The condenser temperature at EPU is about 105°F. Hence summer conditions at EPU will be similar to the conditions for the CLB at nominal power. |
| Hotwell temperature after heater bypass | N/A | 140°F | Conservative assumption |
| SRV Cycling (upper) setpoint | 1080 psid to 1047.6 psid (between RPV and Drywell) | 1080 psid to 1047.6 psid (between RPV and Drywell) | Nominal setpoints |

NON-PROPRIETARY INFORMATION

**Table SPSB-C-14-2**
**ECCS Initial Conditions and Parameters**

| Parameter | CPPU Nominal | CPPU Analysis Value | Comments |
|---|---|---|---|
| Core Spray Flow | Curve of flow vs. vessel-suppression pool $\Delta P$ | Same as nominal. | The core spray system will be used for level control only after the feedwater is depleted.<br><br>During alternate shutdown cooling a flow of 3,500 gpm at 100 psid was used. |
| RHR Flow ($t > 600$ seconds) | 7,000 gpm | 7,000 gpm | The Appendix R analysis is performed assuming nominal conditions.<br><br>Based on the results of the analysis, the available RHR pump can be split between vessel and suppression pool cooling. |
| RHR Hx Tube Plugging | N/A | 5% | |
| RHRSW Flow | 2,950-3,140 gpm | 2,950 gpm | The Appendix R analysis is performed assuming nominal conditions. |
| RHRSW Inlet Temperature | 32-85°F | 85.1°F | Maximum allowable service water temperature |

**Table SPSB-C-14-3**
**Primary Containment Initial Conditions and Parameters**

| Parameter | CPPU Nominal Value | CPPU Analysis Value | Comments |
|---|---|---|---|
| Drywell Temperature | 110-170°F | 170°F | The highest drywell temperature is used. (Not a major impact on suppression pool temperature due to the low heat capacity of nitrogen) |
| Drywell Pressure | 16.4 psia | 16.4 psia | Minimal to no impact on the Appendix R suppression pool temperature |
| Suppression Pool Temperature | 88°F | 90°F | A 2°F uncertainty is applied to account for instrument uncertainty |
| Suppression pool Pressure | 14.7 psia | 14.7 psia | |
| Drywell Humidity | 20 -100 % | 100% | Maximum drywell humidity consistent with CLB analysis |
| Suppression Pool Humidity | 100% | 100% | Minimal to no impact on the Appendix R suppression pool temperature |
| Suppression Pool Water Volume | 68,000- 70,000 $ft^3$ | 68,000 $ft^3$ | Technical Specification minimum |
| Drywell Free Volume | 128,370 -131,470 $ft^3$ | 131,470 $ft^3$ (includes vents) | The maximum value in OPL-4A is used for small and intermediate break LOCAs and small steam breaks |
| Suppression Pool Total Volume | For the minimum water volume of 68,000 $ft^3$, the suppression pool free volume is 107,104.8 $ft^3$ for delta P = 0.0 and 105,932.0 $ft^3$ for delta P > 0.0 where delta P is the pressure difference between drywell and suppression pool. | 173,932 $ft^3$ | The value at delta P > 0 of 105,932.0 $ft^3$ is used for a total volume of 105,932 + 68,000 = 173,932 $ft^3$ |

NON-PROPRIETARY INFORMATION

**Table SPSB-C-14-4**
**CPPU Limiting Case Timeline**

| Time (sec) | EVENT |
|---|---|
| 0 | Reactor scram, MSIVs start to close |
| > 0 | Reactor vessel water level maintained with feedwater |
| 600 | Suppression pool cooling initiated |
| 1,800 | Feedwater heaters bypassed |
| 3,600 | Controlled cooldown initiated at 100°F/hr |
| ~10,800 | Cooldown complete, reactor pressure stabilized at ~100 psig, level maintained by core spray when feedwater depleted |
| ~14,400 | Initiate alternate shutdown cooling with one core spray pump |

Part (b):

Review of the GOTHIC 7.0 SE for the Kewaunee Nuclear Power Plant

The NRC safety evaluation (SE), dated September 29, 2003, for the application of GOTHIC 7.0 at the Kewaunee Nuclear Power Plant has been reviewed. It appears from the SE that Kewaunee was proposing to use some specific features of GOTHIC 7.0 that were not included in previous versions of GOTHIC. Accordingly, the NRC acceptance was predicated on the following conditions, which are stated in the SE:

1. The height effect scaling factor $\lambda_h$ applied to the heat and mass transfer analogy, shall not be used for the Kewaunee licensing calculations.
2. The Gido-Koestel (G-K) correlation shall not be used for Kewaunee licensing calculations.
3. The inclusion of mist in the mist diffusion layer model (MDLM) shall not be used for Kewaunee licensing calculations.

In addition, for the Kewaunee application, the SE stated:

1. It is not necessary to apply the proposed bias term to the mist diffusion layer model for Kewaunee licensing calculations.
2. It is not necessary to use a combination of Uchida and MDLM for the containment heat structures. MDLM may be used for heat transfer to all structures for Kewaunee licensing calculations.

The VYNPS application of GOTHIC 7.0 for the Appendix R and station blackout (SBO) suppression pool temperature analyses are consistent with the above conditions and limitations specified in the Kewaunee SE. The VYNPS application of GOTHIC 7.0 was based on maintaining the latest version of the software and did not invoke new code options or enhancements. The current licensing basis Appendix R and SBO suppression pool analyses were updated with GOTHIC 7.0 models for the suppression pool temperature evaluations under EPU conditions.

The GOTHIC model of the VYNPS containment for EPU conditions does not model heat losses to the reactor building from primary containment (i.e., either the drywell or suppression pool chamber). The primary containment passive structures, such as the biological shield, drywell shell steel, structural components and the base concrete pad, are not modeled as heat transfer structures. This is a conservative representation of the primary containment and leads to higher suppression pool temperatures.

Two passive heat slabs model the suppression pool chamber wall, one in the liquid space and the other in the vapor space. The heat transfer area for each heat slab is based on approximately one half of the total interior surface area of the suppression pool chamber. The heat transfer coefficient for the heat slab exposed to steam uses a GOTHIC correlation set for turbulent natural convection for face-down geometry. This is an adequate approximation for the top half of the toroidal shape of the suppression pool chamber. The heat transfer from the liquid in the suppression pool to or from the suppression pool wall structure uses a GOTHIC correlation set based on turbulent natural convection for face-up geometry.

The heat transfer on the reactor building side of the suppression pool chamber wall uses a fixed heat transfer coefficient of 0.5 Btu/hr-ft$^{2}$-°F with reactor building temperature of 135°F. The Uchida correlation is not used for VYNPS suppression pool temperature analyses.

Part (c):

Other Uses of GOTHIC 7.0 Supporting EPU

The GOTHIC 7.0 code was also applied in the SBO suppression pool temperature calculation, the post-loss of coolant accident (LOCA) reactor building heat-up calculation and the ECCS corner room heat-up calculation. The SBO GOTHIC model is similar to the Appendix R model with differences limited to the scenarios. The post-LOCA reactor building heat-up model is a compartmental model. A summary of these calculations are provided in this response.

SBO Suppression Pool Temperature Analysis Summary

For the CLB, the GOTHIC code was used for calculating the transient suppression pool temperature for the limiting SBO scenarios. The GOTHIC model cases were updated with new decay heat functions to reflect the EPU conditions. In addition, the model was updated slightly to incorporate as-built values. The peak suppression pool temperature for the EPU SBO limiting case is 187.9°F at 23,880 seconds.

SBO Scenario Description, Assumptions, Inputs and Initial Conditions for Limiting Case

The limiting case SBO scenario for suppression pool temperature was analyzed for EPU conditions. The SBO scenario postulates a complete loss of onsite and offsite AC power.

The vessel is assumed to be isolated at the start of the event. The scenario is modeled as follows:

1. Scram occurs at time zero.
2. High pressure makeup is assumed to be available from HPCI and/or RCIC.
3. Vessel pressure is maintained by SRV cycling.
4. At one hour, sufficient AC power is assumed to be restored such that suppression pool cooling is initiated.
5. A vessel cooldown at 45°F/hr is initiated at one hour. The cooldown proceeds until the SRV low mechanical setpoint is reached.
6. The transient continues with the vessel pressure maintained on the SRV mechanical setpoints until the torus peak temperature is reached.

The key parameters and input assumptions are presented below:

1. The plant is assumed to operate at 20% over CLB power, and an additional 2% calorimetric uncertainty is used in the analysis. This results in an assumed power of 1950 MWt.
2. The current licensing basis GOTHIC reactor vessel model assumes a full core of GE-9 fuel bundles. It was demonstrated that the stored energy in the reactor vessel for a full core of GE-9 fuel bundles bounds a full core of GE-14 fuel bundles at EPU conditions.
3. The ANSI/ANS 5.1-1979 decay heat with a 2-sigma uncertainty is used in the analysis.
4. The following tables provide the initial conditions used in the calculation. Table SPSB-C-14-5 presents the reactor vessel and core initial conditions; Table SPSB-C-14-6 presents the ECCS initial conditions and parameters; Table SPSB-C-14-7 presents the primary containment initial conditions; and Table SPSB-C-14-8 summarizes the timing of the limiting case.

**Table SPSB-C-14-5**
**CPPU SBO Vessel and Core Initial Conditions**

| Parameter | CPPU Nominal Value | CPPU Analysis Value | Comments |
|---|---|---|---|
| Power | 1912 MWt | 1950 MWt | EPU plus 2% uncertainty |
| Decay Heat | ANS 5.1 | ANS 5.1 +2-sigma | |
| Vessel Pressure | 1020 psia | 1045 psia | The use of 1045 psia for the vessel pressure for the suppression pool temperature Appendix R analysis is conservative. |
| Vessel Level | 162 inches | 172 inches | Analysis value conservatively accounts for 3 inches above normal water level (uncertainty and operational fluctuations) and 7 inches for dimensional uncertainties. |
| MSIV closure time | 3-5 seconds | 3 seconds | The Technical Specification minimum value closure retains more energy in the vessel |
| Core Flow Rate | 48.0E6 lb/hr | 51.36E6 lb/hr | Increased core flow (ICF) conditions lead to higher vessel stored energy. |
| Initial Feedwater Flowrate | 7.88E6 lb/hr | 0 lb/hr | The feedwater is lost due to SBO. |
| SRV Cycling (upper) setpoint | 1080 psid to 1047.6 psid (between RPV and drywell) | 1080 psid to 1047.6 psid (between RPV and drywell) | Nominal setpoints |

NON-PROPRIETARY INFORMATION

**Table SPSB-C-14-6**
**CPPU SBO ECCS Initial Conditions and Parameters**

| Parameter | CPPU Nominal | CPPU Analysis Value | Comments |
|---|---|---|---|
| Core Spray Flow | Curve of flow vs. vessel-suppression pool ΔP. | N/A | Inventory is maintained with HPCI or/and RCIC. |
| RHR Flow (t > 3600 sec.) | 7,000 gpm | 6,400 gpm | The SBO analysis is performed assuming conservative LOCA DBA conditions.<br><br>The RHR pump is used for suppression pool cooling. |
| RHR Hx Tube Plugging | N/A | 5% | |
| RHRSW Flow | 2,950-3,140 gpm | 2,700 gpm | The SBO analysis is performed assuming conservative LOCA DBA conditions. |
| RHRSW Inlet Temperature | 32-85°F | 85.0°F | Maximum allowable service water temperature |

**Table SPSB-C-14-7**
**CPPU SBO Primary Containment Initial Conditions and Parameters**

| Parameter | CPPU Nominal Value | CPPU Analysis Value | Comments |
|---|---|---|---|
| Drywell Temperature | 110-170°F | 170°F | The highest drywell temperature is used. (Not a major impact on suppression pool temperature due to the low heat capacity of nitrogen.) |
| Drywell Pressure | 16.4 psia | 16.4 psia | Minimal to no impact on the SBO suppression pool temperature. |
| Suppression Pool Temperature | 88°F | 90°F | A 2°F uncertainty is applied to account for instrument uncertainty. |
| Suppression pool Pressure | 14.7 psia | 14.7 psia | |
| Drywell Humidity | 20 -100 % | 100% | Maximum drywell humidity consistent with CLB analysis. |
| Suppression Pool Humidity | 100% | 100% | Minimal to no impact on the SBO suppression pool temperature. |
| Suppression Pool Water Volume | 68,000- 70,000ft$^3$ | 68,000 ft$^3$ | Technical Specification minimum |
| Drywell Free Volume | 128,370 -131,470 ft$^3$ | 131,470 ft$^3$ (includes vent air volume) | The maximum value in OPL-4A is used for small and intermediate break LOCAs and small steam breaks. (115139.3 + 16703.0 – 372.3 (part of vacuum breaker line)) = 131,470 ft$^3$ GOTHIC input adds part of the vacuum breaker piping to the drywell |
| Suppression Pool Total Volume | For the minimum water volume of 68,000 ft$^3$, the suppression pool free volume is 107,104.8 ft$^3$ for delta P = 0.0 and 105,932.0 ft$^3$ for delta P > 0.0 where delta P is the pressure difference between drywell and suppression pool. | 174,031.4 ft$^3$ | The value at delta P > 0 of 105,932.0 ft$^3$ is used for a total volume of 105,932 + 68,000 + 99.4 (vacuum breaker piping) = 174,031.4 ft$^3$ |

**Table SPSB-C-14-8**
**CPPU SBO Limiting Case Timeline**

| Time (sec) | EVENT |
|---|---|
| 0 | Reactor scram, MSIVs start to close |
| > 0 | Reactor vessel water level maintained with HPCI and/or RCIC |
| > 0 | Vessel pressure is maintained by SRV cycling |
| 3,600 | Vessel cooldown at 45°F/hr is initiated |
| 3,600 | RHR suppression pool cooling initiated |
| ~21,600 | Cooldown complete, reactor pressure stabilized at ~100 psig and maintained on the SRV mechanical setpoints (until the torus peak temperature is reached). Level maintained with HPCI and/or RCIC. |
| ~23,900 | Peak suppression pool temperature reached |

Post LOCA Reactor Building Temperature Analysis Summary

The VYNPS reactor building GOTHIC model for the post-LOCA heat-up analysis consists of 19 volumes, 438 flow paths, 2 heater components, 6 volumetric fan components, 59 thermal conductors, 15 thermal conductor heat transfer coefficient (HTC) types. Heat loads in the reactor building consist of the drywell, suppression pool, spent fuel pool, emergency core cooling system and solar loads. The post LOCA reactor building heat-up is affected by EPU through the increase in suppression pool temperature. The drywell temperature profile used in the CLB analysis did not have to be changed since it bounded the EPU drywell temperature profile. Other heat-load assumptions in the CLB analysis were not affected or bounded EPU conditions. As a result, the only change to the CLB GOTHIC model was to update the suppression pool temperature.

**Table SPSB-C-14-9**
**CPPU Post-LOCA Reactor Building Heat-up Analysis Inputs**

| Reactor Building Parameter | Initial Condition | Comment |
|---|---|---|
| Initial Temperature | 100°F | CLB assumption |
| Relative Humidity | 60% | CLB |
| Peak Spent Fuel Pool | 150°F | CLB |
| Peak Suppression Pool Temperature | 195°F | Updated for EPU |
| Peak Drywell Temperature | 345°F | CLB and bounds EPU |

The GOTHIC reactor building model was reviewed considering the NRC safety evaluation (SE), dated September 29, 2003, for the application of GOTHIC 7.0 at the Kewaunee Nuclear Power Plant. Some heat conductor heat transfer coefficients used in the analysis have the "condensation option" for vertical and horizontal surfaces set to "MAX". This condensation option will use the maximum value obtained from the Uchida or Gido-

Koestel correlations. The NRC SE on GOTHIC 7.0 for Kewaunee stated that Gido-Koestel shall not be used for DBA analyses.

The reactor building post-LOCA calculation was re-run with the HTC condensation option set to "UCHIDA" for all heat conductors which use the "MAX"" condensation option. The post-LOCA reactor building results did not change when these options were changed. This is expected because the post-LOCA reactor building heat-up calculation does not involve a two-phase mixture from a high energy line break. Therefore, condensation has a negligible effect on the heat transfer calculations.

ECCS Corner Rooms Evaluation

There are four corner rooms in the Mark-I secondary containment structure. Two of these rooms at VYNPS, northeast and southeast, contain safety related ECCS equipment that is currently cooled using air coolers with cooling water supplied by the service water system. Each corner room is divided into two levels, an upper corner room (232' elevation) and a lower corner room (213' elevation). The GOTHIC code is used to calculate the corner room temperatures (upper and lower room) for one year, post LOCA. These results are used for equipment qualification. All the heat slabs in the corner rooms are modeled. Neither Uchida nor Gido-Koestel correlations are used. Table SPSB-C-14-10 summarizes the results.

**Table SPSB-C-14-10**
**ECCS Corner Room Temperatures (post-DBA LOCA)**

| Case No. | Duration at this temperature (month) | Temperature at Elevation 213 (°F) | Temperature at Elevation 232 (°F) | Change from Current Licensed Power (CLTP) |
|---|---|---|---|---|
| 1 | 1 | Max Temperature 159.0 | Max Temperature 159.8 | Elevation 213 +4.0°F Elevation 232 +6.4°F |
| 2 | 1 | 148.8 | 147.7 | None |
| 3 | 1 | 142.2 | 146.1 | None |
| 4 | 1 | 128.6 | 120.4 | None |
| 5 | 1 | 115.0 | 109.5 | None |
| 6 | 2 | 107.3 | 107.4 | None |
| 7 | 1 | 115.0 | 109.5 | None |
| 8 | 1 | 124.2 | 111.8 | None |
| 9 | 2 | 142.1 | 124.3 | None |
| 10 | 1 | 143.6 | 125.0 | None |

Part (d):

10CFR50.59 Evaluation

The VYNPS application for a license amendment for EPU includes the use of GOTHIC 7.0. Since this is a request for a change to the licensing basis pursuant to 10CFR50.90, NRC regulation 10CFR50.59 is not applicable.

Part (e):

Generic letter (GL) 83-11 and GL 83-11 Supplement 1

VY has followed the guidance in GL 83-11 and GL 83-11 Supplement 1. The GOTHIC code was developed by Numerical Applications, Incorporated (NAI) and is distributed by the Electric Power Research Institute (EPRI). Entergy is a member organization in the EPRI GOTHIC user group and VY obtained GOTHIC 7.0 from EPRI as a safety related code. Safety-related application of the GOTHIC 7.0 code is controlled by Entergy and VY software quality assurance (SQA) procedures. The GOTHIC code has been benchmarked as discussed in the response to RAI No. 14(f) below. As a member of the EPRI GOTHIC user group, Entergy and VY receive code updates and applicable condition reports. GOTHIC condition reports are entered into the VYNPS corrective actions program. GOTHIC problem summaries in these reports are evaluated and the dispositions are documented in a safety related evaluation. The evaluation may include discussions with other Entergy GOTHIC users. VY personnel who use the GOTHIC code are limited to those individuals trained by the code developer or were trained by someone that was trained by the code developer, and have considerable experience in the application of GOTHIC for containment response calculations and high energy line break (HELB) evaluations. Other Entergy GOTHIC users are available as a resource to VY. At least one qualified Entergy GOTHIC user normally attends the user group meetings. Proceedings and materials from user group meetings are shared among all the Entergy qualified GOTHIC users.

Part (f):

GOTHIC 7.0 Benchmarking

Acceptance testing (validation and verification) of the installation of the GOTHIC 7.0 code was performed in accordance with Entergy and VY SQA requirements and is documented in a safety related calculation. The acceptance testing process followed the steps outlined in the GOTHIC code "Installation and Operations Manual," Version 7.0. Various sample problems (test problem models) are provided by the code developer to test and verify the user installation. In addition to the code developer recommended sample problems, a VYNPS specific GOTHIC model was also run with GOTHIC 7.0. The model selected was the VYNPS limiting LOCA suppression pool temperature case for analyses performed for VYNPS license amendment 163. This case was originally run using GOTHIC 5.0e. The benchmarking results of GOTHIC 7.0 reproduced the results of GOTHIC 5.0e.

## RAI SPSB-C-15

The application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.1.1.1 (b), discusses an evaluation that was performed regarding the possibility of steam injestion into the ECCS suction strainers. Please provide this evaluation (Reference 26) for NRC review.

## Response to RAI SPSB-C-15

Based on discussions with the NRC staff, it is understood that making a copy of Reference 26 available for inspection at VYNPS is sufficient to satisfy this request.

The referenced evaluation was done as part of the design change that installed larger capacity ECCS suction strainers in response to NRC Bulletin 96-03. The evaluation was not done specifically in support of the power uprate submittal.

The evaluation considers the location and dimensions of the ECCS suction strainers in the torus in relation to the downcomers and the safety/relief valve (SRV) discharge devices (i.e., tee-quenchers). The evaluation also considers both air and steam bubbles from the downcomers and SRVs with the inputs defined by the Mark I containment program load definition report. Power uprate does not affect any of the above physical parameters, and the Mark I containment program load definition report remains applicable. Therefore, the conclusions of the evaluation remain applicable for power uprate.

## RAI SPSB-C-16

The application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.1.2.1, discusses LOCA loads. Explain why vent thrust loads are less than those calculated during the Mark I Containment Long Term Program.

## Response to RAI SPSB-C-16

The vent thrust loads for the Mark I Long-Term Containment Program (LTP) and for the EPU were both calculated using the vent thrust load methodology described in NEDO-21888 using the containment response calculated with M3CPT. The difference in methods between the Mark I LTP and the CPPU analysis is in the calculation of the vessel blowdown break flow rate and enthalpy used in the M3CPT calculation.

For the Mark I LTP analyses, the break flow rates were calculated internally by M3CPT using the vessel blowdown model built into the code. [[

]]

including the time period when the pressure differential between the DW and WW peaks. In general, the maximum vent thrust occurs during this time period.

For the CPPU analyses, the break flow rate and enthalpy are calculated, external to M3CPT, using the LAMB computer code. The LAMB-generated break flow rate and enthalpy histories are input to the M3CPT computer code. [[

]] The resulting lower drywell-to-wetwell pressure differences and vent flow rates produce the lower vent thrust loads obtained in the EPU analysis.

The use of the LAMB code for calculating blowdown flows and enthalpies for use in the M3CPT analyses was identified in ELTR1 (NEDC-32424P) which is referenced in the CPPU LTR (NEDC-33004P) as the basis for the containment evaluations. The M3CPT code itself is still used to calculate the drywell and wetwell pressure and temperature response and vent flow rates.

## RAI SPSB-C-17

The application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.1.2.3, discusses subcompartment pressurization. How was the estimate obtained that the effect of the increase in subcooling would be less than 3 psid on the resulting annulus pressure?

## Response to RAI SPSB-C-17

The increase in subcooling results in a slightly higher break flow rate into the annulus region. The estimates are based on a constant enthalpy process with critical flow at both the break location and the exit of the annulus flow path to the drywell. The increased pressure is based on the pressure that results from the increased flow rate into the annulus region, assuming a quasi-steady state where the flow rate into the annulus equals the flow rate out. This is consistent with the method used in the CLTP calculation.

## RAI SPSB-C-18

The application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.1.5, discusses a hardened wetwell vent system installed at VYNPS in response to GL 89-16. Is the hardened vent capability maintained without any changes in acceptance criteria or analytical methods?

## Response to RAI SPSB-C-18

The vent design criterion was to maintain containment design pressure assuming a steaming rate associated with 1% of full reactor power. The actual capability of the VYNPS design was determined to be equivalent to 1.3% of the current licensed thermal power of 1593 MWt.

Therefore, there is sufficient capacity to accommodate a 20% power increase and still meet the design criterion of 1% of full reactor power.

## RAI SPSB-C-19

The application dated September 10, 2003 (Reference 1), Attachment 6, Table 4-1, provides VYNPS containment performance results. Explain why the use of the CPPU method for the CLTP increases the peak drywell pressure by 3.4 psi and the peak drywell air space temperature by 3.7 °F. Page 4-4 says the Moody slip critical flow model was responsible for most of this increase. What critical flow model was used for the Mark I Long Term Program?

## Response to RAI SPSB-C-19

For the current licensing basis, the peak drywell pressure of 38.2 psig reported in the FSAR was calculated using the M3CPT computer program. The break flows and enthalpies used in M3CPT were calculated by the LAMB code using the Homogeneous Equilibrium Critical Flow Model (HEM). The HEM critical flow model was also used for the Mark I Long Term Program (LTP) M3CPT containment analyses. It is noted that for the Mark I LTP analyses, the HEM model was applied with the M3CPT internal vessel blowdown model (see response to RAI SPSB-C-16).

With the CPPU methodology, the LAMB break flow rates used in M3CPT were based on the Moody Slip Flow critical flow model (SLIP).

The SLIP critical flow model calculates higher break flow rates compared to the HEM model. Therefore, use of the higher LAMB break flow rates in the M3CPT calculation with the CPPU methodology produced a higher peak drywell pressure (41.6 psig) relative to the FSAR value (38.2 psig).

## RAI SPSB-C-20

Verify that the primary containment long-term pressure and temperature responses have been determined using the five cooling conditions listed in Section 5.2.4.3 of the UFSAR.

## Response to RAI SPSB-C-20

Only the most limiting cases were evaluated. Specifically, cases 4 and 5 listed in Section 5.2.4.3 of the UFSAR. The analytical results for these cases for the current licensed power level are shown in UFSAR Section 14.6.3.2.2, where cases 4 and 5 are identified as D and E, respectively.. These cases provide the maximum suppression pool temperatures, and for ECCS NPSH evaluations, the minimum containment overpressure available.

## RAI SPSB-C-21

Verify that the proposed EPU amendment is consistent with the guidance of Regulatory Guide (RG) 1.82 Revision 3. In addition, confirm that RG 1.82 Revision 3, or at least Section 2.1, will become part of the VYNPS licensing basis if the proposed amendment is approved.

## Response to RAI SPSB-C-21

Regulatory Guide 1.82 Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant-Accident" was issued in November 2003, after the EPU license amendment application was made. As discussed during a telecon with the NRC staff on June 7, 2004, VY agreed to address the aspects of RG 1.82 that are pertinent to crediting containment overpressure. The response addresses the regulatory positions in Section 2.1.1, "Net Positive Suction head of ECCS and Containment Heat Removal Pumps." It is not Entergy's intention that RG 1.82 Revision 3, Section 2.1 will become part of the VYNPS licensing basis. However, as discussed below, VYNPS meets the intent of RG 1.82 Revision 3, Section 2.1.

Regulatory Guide 1.82, Revision 3, Position 2.1.1.1 addresses the current licensing basis condition where no increases in containment pressure from that present prior to the postulated LOCA is assumed. Section 2.1 of the guide addresses the features needed to minimize the potential for loss of net positive suction head (NPSH). Regulatory Guide Position 2.1.1.2 addresses containment overpressure. Specifically, Regulatory Position 2.1.1.2 states the following:

> *"For certain operating BWRs for which the design cannot be practicably altered, conformance with Regulatory Position 2.1.1.1 may not be possible. In these cases, no additional containment pressure should be included in the determination of available NPSH than is necessary to preclude pump cavitation. Calculation of available containment pressure should underestimate the expected containment pressure when determining available NPSH for this situation. Calculation of suppression pool water temperature should overestimate the expected temperature when determining available NPSH."*

The VYNPS NPSH calculations for EPU conditions meet the intent of Regulatory Position 2.1.1.2. The containment pressure required for NPSH is less than the calculated available post-accident containment overpressure. This is demonstrated in Figure 4-6 of Attachment 6 to the application dated September 10, 2003. Figure 4-6 shows approximately 1.5 psi margin between the stepped overpressure credit and the overpressure available. The calculation of available overpressure underestimates the expected overpressure, and suppression pool temperature calculations are performed with conservative assumptions to overestimate the expected temperature. The response to RAI SPSB-C-11 discusses the conservatisms applied to the calculations.

Regulatory Position 2.1.1.3 addresses crediting pump operation in the cavitation mode. The containment overpressure credit is based on meeting the pump manufacturer's recommended minimum available NPSH requirements. The pump manufacturer's recommended minimum available NPSH is based on assuring acceptable pump performance and reliability. See the response to RAI SPSB-C-25 for additional information on pump performance and cavitation. The intent of Regulatory Position 2.1.1.3 is met by meeting the minimum NPSH requirement.

The treatment of decay and residual heat meets the intent of Regulatory Position 2.1.1.4. The calculations assume a 2% calorimetric uncertainty and the decay heat calculation also applies the ANSI/ANS 5.1-1979 2-sigma uncertainty. The decay heat assumptions are also discussed in the responses to RAIs SPSB-C-4 and SPSB-C-5.

Regulatory Position 2.1.1.5 states that the hot channel correlation factor specified in ANSI/HI 1.1-1.5-1994 should not be used. VYNPS has not applied this correlation in determining the margin between the available and required NPSH.

The initial suppression pool volume assumed in the suppression pool temperature and NPSH calculations is the Technical Specification minimum value. This assumption meets the intent of Regulatory Position 2.1.1.6.

Piping losses in the pump suction that affect the NPSH have been included in the calculation. This meets the intent of Regulatory Position 2.1.1.7. The response to RAI SPSB-C-26 includes the NPSH calculation.

The NPSH treatment of suction strainer screen debris loading meets the intent of Regulatory Position 2.1.1.8. The response to RAI SPSB-C-6 notes that there have been no changes adversely affecting debris loading since the completion of actions requested by NRC Bulletin 96-03.

Regulatory Position 2.1.1.9 states that NPSH calculations should be performed as a function of time until it is clear that the available NPSH will not decrease further. The NPSH calculation meets the intent of this regulatory position. Revised Figure 4-6, provided with the response to RAI SPSB-C-23, shows the time dependent NPSH requirements. The figure shows that NPSH will not decrease further as demonstrated by the decreasing amount of overpressure credit required after approximately eight hours.

## RAI SPSB-C-22

Describe how the VYNPS emergency operating procedures will be revised to ensure that the containment accident pressure will be prevented from falling below the pressure required for adequate available NPSH.

## Response to RAI SPSB-C-22

The VYNPS emergency operating procedures (EOPs) do not require revision to ensure that the containment accident pressure will be prevented from falling below the pressure required for adequate available NPSH. Current EOPs incorporate guidance to ensure that containment accident pressure will be prevented from falling below the pressure required for adequate available NPSH.

Per VYNPS emergency operating procedure (EOP) EOP-1, "RPV Control," after an automatic action level has been reached, operators are directed to verify applicable automatic actions have occurred. Verifying automatic actions provides backup confirmation that all isolation valves have closed on a primary containment isolation signal.

VYNPS EOPs establish NPSH limits for residual heat removal (RHR) and core spray (CS) pumps. (Separate limits are provided for RHR and CS). The NPSH limit is a function of pump flow, torus water temperature, and suppression chamber pressure. It is used to preclude ECCS pump damage due to cavitation and to ensure adequate coolant flow. As overpressure increases, the static pressure and margin to saturation at the pump inlet also increase. The available NPSH therefore increases with overpressure.

In accordance with EOP-1, when using RHR for an injection system, operators are directed to inject through the heat exchanger as soon as possible and to control and maintain pump flow below the RHR NPSH Limit. For the core spray system, operators are directed to control and maintain pump flow below the CS NPSH Limit.

EOP-3, "Primary Containment Control," Note 5 states: "Reducing primary containment pressure will reduce the available NPSH for pumps taking suction from the torus." Per the EOP Study Guide, if there is no future need for sprays and containment overpressure is desired to provide adequate NPSH for pumps drawing suction from the suppression pool, sprays may be terminated at a higher pressure.

In accordance with EOP-3, drywell sprays are initiated before containment temperature reaches 280 °F or when torus pressure exceeds 10 psi. Containment sprays should isolate automatically when drywell pressure decreases to 2.5 psig. Both of these steps in EOP-3 provide reference to Caution #5 emphasizing the relationship between primary containment pressure and available NPSH.

Also, per EOP-3, once the high drywell pressure isolation occurs, containment venting is directed only after a reactor pressure vessel emergency depressurization (RPV-ED) is required and prior to exceeding the primary containment pressure limit (PCPL-A curve in EOP-3). In the event that containment venting is required, operators will vent the containment to control pressure below the PCPL-A curve. The pressure at which containment is maintained during venting is based on considerations of NPSH for the RHR and core spray pumps, expected release rates, and total releases. Therefore, sufficient containment overpressure is preserved.

## RAI SPSB-C-23

The application dated September 10, 2003 (Reference 1), Attachment 6, Table 4-2 and Figure 4-6, show that the containment accident pressure requested for adequate available NPSH is 1.3 psig at 50 hours. When is containment accident pressure no longer required?

## Response to RAI SPSB-C-23

The analysis run time was extended and the results show that the time when containment accident pressure is no longer required is 200,000 seconds, or 55.6 hours. The analysis was also revised to account for an increase in the assumed containment leakage rate from 0.8 % per day to 1.5 % per day. This is a conservative adjustment that slightly reduces the available containment overpressure. The adjustment was made in order to be consistent with the assumptions in VY's July 31, 2003, license amendment request for alternative source term.

Revised Table 4-2 and Figure 4-6 are enclosed. The overpressure credit requested has not changed. The time after LOCA has been extended from 180,000 seconds to 200,000 seconds.

NON-PROPRIETARY INFORMATION

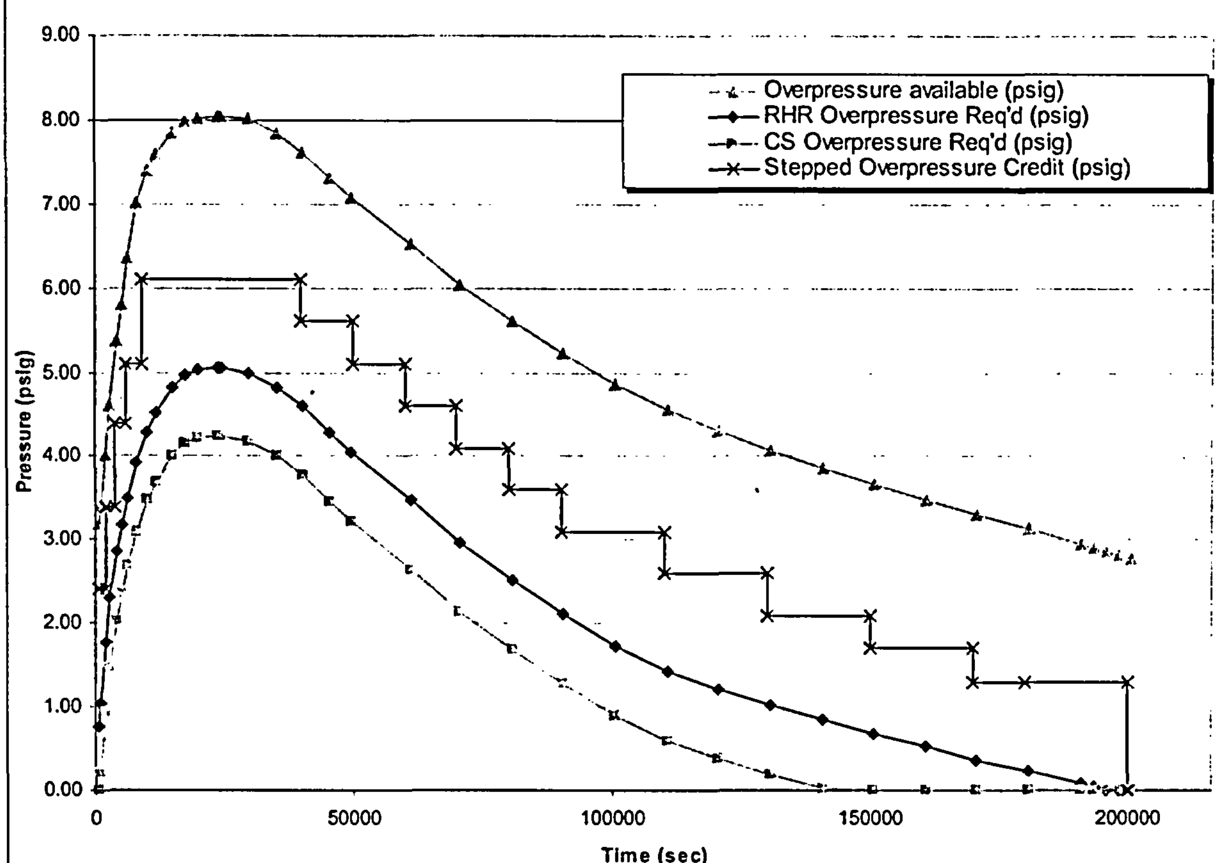
## Table 4-2 (Revised)

## VYNPS Overpressure Credit for NPSH DBA LOCA-Long Term

| Time After LOCA (sec) | Overpressure Credit (psig) |
|---|---|
| 600 | 2.4 |
| 2,000 | 2.4 |
| 2,001 | 3.4 |
| 4,000 | 3.4 |
| 4,001 | 4.4 |
| 6,000 | 4.4 |
| 6,001 | 5.1 |
| 9,000 | 5.1 |
| 9,001 | 6.1 |
| 40,000 | 6.1 |
| 40,001 | 5.6 |
| 50,000 | 5.6 |
| 50,001 | 5.1 |
| 60,000 | 5.1 |
| 60,001 | 4.6 |
| 70,000 | 4.6 |
| 70,001 | 4.1 |
| 80,000 | 4.1 |
| 80,001 | 3.6 |
| 90,000 | 3.6 |
| 90,001 | 3.1 |
| 110,000 | 3.1 |
| 110,001 | 2.6 |
| 130,000 | 2.6 |
| 130,001 | 2.1 |
| 150,000 | 2.1 |
| 150,001 | 1.7 |
| 170,000 | 1.7 |
| 170,001 | 1.3 |
| 180,000 | 1.3 |
| 200,000 | 1.3 |
| 200,001 | 0 |

NON-PROPRIETARY INFORMATION

Figure 4-6 (Revised)
Overpressure Required for NPSH DBA LOCA–Long Term




## RAI SPSB-C-24

Does the higher temperature of the drywell air following a LOCA for the CPPU, compared to the calculated drywell air temperature of the CLTP, affect drywell bypass considerations in any way? Please explain.

## Response to RAI SPSB-C-24

According to the Vermont Yankee Nuclear Power Station (VYNPS) Technical Specifications (TS), VYNPS has a surveillance requirement to demonstrate that the drywell-to-wetwell leakage shall not exceed the equivalent of the leakage rate through a 1-inch orifice. This is consistent with the generic acceptance criterion for Mark I plants specified in Section 6.2.1 of the Standard Review Plan (SRP, NUREG-0800).

The limiting postulated event for drywell-to-wetwell bypass leakage is not the DBA-LOCA. Therefore changes in the DBA-LOCA pressure or temperature response do not impact bypass leakage requirements. The maximum bypass leakage will occur for a break size that maintains a drywell-to-suppression chamber pressure difference that is just less than that required to clear the drywell vents and for the longest credible duration. For this limiting break size, i.e., the break size with the minimum associated allowable leakage area, sufficient break flow is injected into the drywell to maintain a steady pressure difference between the drywell and suppression chamber while not clearing the drywell vent. [[

]]

Since the primary factors ([[ ]]) affecting the peak containment pressure during steam bypass events are not adversely impacted by power uprate, the existing criteria for drywell bypass leakage for VYNPS remain applicable at the uprated power conditions.

Therefore, since steam bypass for the limiting condition is independent of reactor power level, it is not adversely impacted by power uprate and the existing criteria for drywell bypass leakage for VYNPS remain applicable at uprated power conditions.

## RAI SPSB-C-25

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, the Hydraulic Institute recommends margin above the required NPSH to suppress cavitation. What margin is needed for the VYNPS pumps crediting containment accident pressure and how is this margin accounted for in the VYNPS NPSH calculations? Provide quantitative information.

## Response to RAI SPSB-C-25

The required NPSH ($NPSH_R$) information provided for the Vermont Yankee Nuclear Power Station (VYNPS) core spray (CS) and residual heat removal (RHR) pumps by the manufacturer specifically address time-phased operational requirements with low available NPSH ($NPSH_A$). No specific margin is included or required in the $NPSH_A$ calculation. However, there is some margin between the overpressure required and the credited overpressure requested and more margin to the overpressure available.

The following general discussion provides additional background information regarding the topic of NPSH margin for pumps:

> The two primary bases for requiring levels of $NPSH_A$ above $NPSH_R$ are hydraulic performance and mechanical reliability. By meeting or exceeding the $NPSH_R$ for a particular flow or range of flows, hydraulic performance is maintained and mechanical reliability is assured for extended operation.
>
> Hydraulic performance can be reduced below the non-cavitating performance curve with reduced margins of NPSH. This degradation is typically less than margins provided for in the sizing of a pump to deliver its design performance.
>
> For a given pump design, the mechanical impact to impeller surfaces and other parts of the pump due to cavitation is determined by the frequency of such operation, the duration and the severity of the event(s), as well as material durability. Typically, all pumps are exposed to brief periods of cavitation during startup or other major system upsets with little, if any, measurable impact.
>
> Pumps installed in safety systems are fitted with materials of construction and mechanical parts that are qualified for extensive operating periods and frequent cyclic operation well beyond their expected service life.
>
> Although certain safety-related pumps can be described as having moderate suction energy levels, the frequency and duration of the events when $NPSH_A$ levels are at or near defined $NPSH_R$ levels, are minor when compared to the long-term design qualification of the pump.

## RAI SPSB-C-26

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, please provide for NRC review the VYNPS calculations of NPSH and containment accident pressure associated with the EPU amendment request.

## Response to RAI SPSB-C-26

The NPSH calculations are documented in VYNPS calculations VYC-0808, Rev. 6, CCN05 and VYC-2314, Rev. 0 and are included in Attachment 4 of this submittal as Exhibits 1 and 2, respectively. Based on discussions with NRC staff, it is understood that providing these calculations should be sufficient for NRC staff review needs.

The calculation of containment accident pressure, used as input to the LOCA NPSH calculation, was performed by GE.

## RAI SPSB-C-27

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, provide the results of an analysis of the stuck open reactor vessel relief valve which demonstrates that adequate NPSH is available for successful operation of the ECCS pumps.

## Response to RAI SPSB-C-27

This event assumes only one of two RHR heat exchangers is available for suppression pool cooling. Operators control reactor inventory with feedwater, and control the cooldown to 80 °F/hr, using turbine bypass valves to direct steam to the main condenser as required. When the reactor pressure permissive allows it, the RHR system is realigned from the suppression pool cooling mode and placed in the shutdown cooling mode. It is conservatively assumed to take 76 minutes to make this transition, during which time the suppression pool is not cooled by the RHR system. The peak suppression pool temperature for this scenario is 182.1 °F.

An alternative scenario was also evaluated, where the RHR system would be allowed to continuously operate in the suppression pool cooling mode instead of transitioning to the shutdown cooling mode, resulted in a peak suppression pool temperature of 177.1 °F.

An NPSH calculation was not performed for the SORV event. However, because the peak suppression pool temperatures shown above are less than 182.6 °F, which is the peak suppression pool temperature for the current design basis LOCA, available NPSH is estimated to be adequate for successful operation of the ECCS pumps.

## Reactor Systems Branch (SRXB)

### RAI SRXB-A-1

Supplement 4 (Reference 5) provides information as to the method used by the licensee to provide oversight of engineering products of GE Nuclear Engineering (GENE) and the licensee's confirmation process related to the GENE analyses. Attachment 1 cites two assessments performed by the licensee at GENE offices during May and October of 2003.

a) Please describe the power uprate confirmation process used by VYNPS, citing documentation and references as appropriate.

b) Please cite the reference for the GENE Quality Assurance Program (QAP) that was used for the VYNPS EPU safety analyses discussed in the power uprate safety analysis report (PUSAR).. Is this QAP also applicable to work performed by Global Nuclear Fuel (GNF) and GE Energy Services (GEES)?

c) Please cite the reference for the VYNPS QAP that was used for the EPU safety analyses oversight. Is this QAP also applicable to the VYNPS "control of off-site services process" that is cited.

d) How are the assessments for the May and October 2003 trips documented, and where is this documentation available? Was there an audit plan, and how was the success of the assessment judged?

e) The summary of the VYNPS confirmation mentions feedback to the GENE performers of comments and resolution. How is this documented? Will the final Design Record Files show the results of the resolution?

f) Please provide further description of any additional assessments planned and the schedule for accomplishing them.

### Response to RAI SRXB-A-1

a) In addition to VYNPS and Entergy procedures (e.g., QA record retention, control of contracted services), the power uprate project utilizes project specific Project Instructions (PI) to direct the complete process including confirmation. The PIs specific to the confirmation process are: EPU-PI-01, "GE Document Reviews, Impact Identification and Documentation"; EPU-PI-12, "Stone and Webster Document Reviews, Impact Identification and Documentation"; and EPU-PI-13, "Vermont Yankee Task Scoping Document Development, Impact Identification and Documentation." These PIs are available at VYNPS, within the project electronic files, and are available for audit.

b) The reference for the GENE Quality Assurance Program (QAP) that was used in the VYNPS EPU safety analyses discussed in the PUSAR is NEDO-11209-04A, Revision 8, dated March 31, 1989, "GE Nuclear Energy Quality Assurance Program Description."

The QAP described in NEDO-11209-04A, Revision 8 is applicable to the work performed by Global Nuclear Fuel (GNF) and GE Energy Services (GEES) for safety analyses performed by these organizations in support of the VYNPS PUSAR.

c) The QA program used for the EPU safety analysis oversight was originally the Vermont Yankee Operational Quality Assurance Manual (VOQAM). As part of an Entergy fleet wide transition, the governing QA program is now the Entergy Quality Assurance Program Manual (QAPM). For VYNPS, this transition occurred in June 2003, and the VOQAM was revised to reference the comparable Entergy QAPM sections that establish the equivalent level of control. The Operational Quality Assurance Program described in the Entergy QAPM is now in effect at VYNPS, and is applicable to all work performed on safety-related structures, systems and components.

   The EPU project has also utilized VYNPS implementing procedures, including AP 0847, "Control of Off-Site Contracted Services," which is a quality-related procedure within the QA program.

d) The May and October assessments are documented in VY Self Assessment Reports and/or Vermont Yankee Quality Assurance Activity-Based Surveillance Reports. Self assessments and QA surveillance reports are available for inspection at VYNPS. The audit plan, and evaluation of the success of the assessment are documented in these reports. Prior to conducting assessments objectives were established, including criteria, to measure the quality of various tasks and activities. Following the conduct of the assessments, the results and demonstration of objectives were documented. This was followed by corrective actions, as necessary.

e) The resolution of all comments made by Entergy concerning GE workscope deliverables in support of the safety analyses discussed in the PUSAR were transmitted back to the VYNPS EPU Project Manager by the GE Nuclear Project Manager as part of the workscope final deliverable for each GE task. These comment resolutions are also contained in the GE design record files for the project.

f) There will be a QA vendor surveillance, as well as another technical assessment, performed this summer (2004) at GE.

## RAI SRXB-A-2

Supplement 4 (reference 5), Attachment 1, Item 1.c discusses dispositions of certain items that have not yet been confirmed since they will be evaluated for the uprated core prior to CPPU implementation. The VYNPS response to this issue, which was raised in the NRC's letter dated December 15, 2003, cites Section 1.1.1 of the GENE CPPU Licensing Topical Report (CLTR), the GENE PUSAR, and Section 1.5 of the NRC's Safety Evaluation for the CLTR, as justifying the assertion that no further analysis is required to be performed for the GENE PUSAR submittal, and that further review of the GNF standard reload analysis methods (GESTAR-II) or the analysis results is not necessary. However, the VYNPS response also notes that the reload licensing analysis (RLA) process is being treated as a design change, requiring formal review and approval of key inputs and output.

previous cycle results and potential core design changes. The results of the meeting are documented in the VYNPS Reload 23 / Cycle 24 design record files 0000-0012-4824 (VYNPS C24 Reload Engineering), 0000-0016-6408 (VYNPS C24 Reload Licensing) and 0000-0016-6409 (VYNPS C24 Reviews).

- Transient selection review meeting – November 6, 2003 – VY and other Entergy personnel participated in this meeting via phone conference with GNF. The transient selection meeting reviews operating plant list (OPL)-3 inputs, equipment out-of-service (OOS) to be analyzed and exposure points to perform the analysis at. The results of the meeting are documented in the VYNPS Reload 23 / Cycle 24 design record files 0000-0012-4824 (VYNPS C24 Reload Engineering), 0000-0016-6408 (VYNPS C24 Reload Licensing) and 0000-0016-6409 (VYNPS C24 Reviews).

- Reload license quality review meeting (mini-review) – January 21, 2004 – VY personnel participated in this meeting at GNF offices and via phone conference call. The reload license quality review meeting reviews the reload analysis and discusses the results with each group who performed the analysis. The results of the meeting are documented in the VYNPS Reload 23 / Cycle 24 design record files 0000-0012-4824 (VYNPS C24 Reload Engineering), 0000-0016-6408 (VYNPS C24 Reload Licensing) and 0000-0016-6409 (VYNPS C24 Reviews).

c) The Supplemental Reload Licensing Report (SRLR) and the Core Operating Limits Report (COLR) are scheduled for completion by November 1, 2004, and December 15, 2004, respectively.

## RAI SRXB-A-3

The disposition of the draft GDC versus final GDC concern was addressed in Supplement 4 (Reference 5), Attachment 4, by providing a revised template SE based on the VYNPS current licensing basis. The revisions correctly note the differences in the draft GDC wording, including the draft use of "acceptable fuel damage limits" versus the final wording of "specified fuel design limits." However, the acronym SAFDLs (specified acceptable fuel design limit(s)) still appears (Sections 2.8.5.3, 2.8.5.4 and 2.8.5.5 for example). Provide a revised template SE that is consistent with the draft GDC wording.

## Response to RAI SRXB-A-3

Exhibit 3 in Attachment 4 to this submittal is a pen-and-ink markup of the Safety Evaluation (SE) template that substitutes the term "acceptable fuel damage limits" (AFDLs) for "specified acceptable fuel design limits" (SAFDLs) to be consistent with the terminology used in the draft General Design Criteria. Based on discussions with the NRC staff, it is understood that providing a markup of the template SE is acceptable to support the NRC staff in making the final changes to the template.

a) Please describe the reload design change process being used, citing documentation and references as appropriate.

b) Please describe the VYNPS participation in the GNF reload design meetings that are cited. What were the dates and how and where are these meetings documented?

c) Please provide the current schedule for the Supplemental Reload Licensing Report and the Core Operating Limits Report.

## Response to RAI SRXB-A-2

a) The current operating cycle (i.e., Cycle 24) reload followed the VYNPS design change process in accordance with plant procedure AP6008. In addition, procedures AP6011 through AP6014 and AP6016 were used in the development, review and approval of reload related design documents. The Cycle 24 core reload design change is documented as VYDC 2003-009.

b) VYNPS personnel participated in several critical design meetings with GNF. These included, but were not limited to, licensing kickoff meeting, fuel bundle design, eigenvalue selection review, transient selection review, and reload license quality review (mini-review). In addition to these reviews, a weekly telephone call was held between the GNF and Entergy personnel to discuss reload related issues.

The dates for some of the following key reviews are:

- General overview meeting – January 20, 2003 – VY and GNF personnel met in Wilmington, NC to discuss VYNPS Cycle 24 reload. The agenda and results of the meeting are available at the VYNPS site in the VYNPS core design folder.

- Licensing kickoff meeting – September 26, 2003 – VY personnel participated in this meeting via phone conference with GNF. The licensing kickoff meeting discusses reload design changes from previous cycle, reload schedule and critical activities. The results of the licensing kickoff meeting are documented in the VYNPS Reload 23 / Cycle 24 design record files 0000-0012-4824 (VYNPS C24 Reload Engineering), 0000-0016-6408 (VYNPS C24 Reload Licensing) and 0000-0016-6409 (VYNPS C24 Reviews).

- Fuel cycle and bundle design – Weekly phone calls between the fuel vendor and VY personnel between March 2003 and July 2003. VY personnel participated in this meeting via phone conference with GNF. The results of these phone calls are summarized in emails located in the VYNPS core design folder.

- Eigenvalue selection review meeting - September 10, 2003 – VY personnel participated in this meeting via phone conference with GNF. The selection meeting discussed hot and cold eigenvalues to be used in the reload design based on

Docket No. 50-271
BVY 04-058

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 8

Extended Power Uprate

Response to Request for Additional Information

Affidavit – General Electric Company

## General Electric Company

## AFFIDAVIT

I, **George B. Stramback**, state as follows:

(1) I am Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Attachment 2 to GE letter GE-VYNPS-AEP-350, Michael Dick (GE) to Craig Nichols (ENOI), *VYNPS Extended Power Uprate - Response to NRC Request for Additional Information, Proprietary and Non-Proprietary Versions,* dated July 1, 2004. The Attachment 2 proprietary information, *GE Responses to NRC RAIs,* is delineated by a double underline inside double square brackets. In each case, the superscript notation[3] refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.790(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a., and (4)b, above.

(5) To address 10 CFR 2.790 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed information in support of NEDC-33090P, *Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate*, Class III (GE Proprietary Information), Revision 0, dated September 2003, which was submitted to the NRC. This power uprate report contains detailed results and conclusions from evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability for the power uprate of a GE BWR, utilizing analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of and applied to perform evaluations of the transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of

the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 1st day of July 2004.

George B. Stramback
George B. Stramback
General Electric Company

Docket No. 50-271
BVY 04-058

Exhibit 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 8

Extended Power Uprate

Response to Request for Additional Information

Markups to RS-001 Safety Evaluation Template

| | |
|---|---|
| MSIVLCS | main steam isolation valve leakage control system |
| MSLB | main steamline break |
| MSSS | main steam supply system |
| MWt | megawatts thermal |
| NEI | Nuclear Energy Institute |
| NPSH | net positive suction head |
| NRC | Nuclear Regulatory Commission |
| NRR | Office of Nuclear Reactor Regulation |
| NSSS | nuclear steam supply system |
| O&M | operations and maintenance |
| P-T | pressure-temperature |
| PWSCC | primary water stress-corrosion cracking |
| RCIC | reactor core isolation cooling |
| RCPB | reactor coolant pressure boundary |
| RCS | reactor coolant system |
| RG | regulatory guide |
| RHR | residual heat removal |
| RS | review standard |
| RWCS | reactor water cleanup system |
| ~~SAFDL~~ | ~~specified~~ acceptable fuel ~~design~~ limit |
| SAG | severe accident guideline |
| SAR | Safety Analysis Report |
| SBO | station blackout |
| SFP | spent fuel pool |
| SFPAVS | spent fuel pool area ventilation system |
| SGTS | standby gas treatment system |
| SLCS | standby liquid control system |
| SRP | Standard Review Plan |
| | |

AFDL
<ORDER ALPHABETICALLY>

damage

## 2.2.3 Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

Reactor pressure vessel internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. The NRC staff reviewed the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with loss-of-coolant accidents (LOCAs), and the identification of design transient occurrences. The NRC staff's review covered (1) the analyses of flow-induced vibration for safety-related and non-safety-related reactor internal components and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also included a comparison of the resulting stresses and CUFs against the corresponding Code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and draft GDC-1, insofar as they require that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a loss of coolant accident; and (4) draft GDC-6, insofar as it requires that the reactor core be designed with appropriate margin to assure that acceptable fuel damage limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5; and other guidance provided in Matrix 2 of RS-001.

(AFDLs)

Technical Evaluation

**[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]**

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of reactor internals and core supports and concludes that the licensee has adequately addressed the effects of the proposed EPU on the reactor internals and core supports. The NRC staff further concludes that the licensee has demonstrated that the reactor internals and core supports will continue to meet the requirements of 10 CFR 50.55a, draft GDC-1, 2, 6, 40, and 42 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the design of the reactor internal and core supports.

## 2.8.5 Accident and Transient Analyses

### 2.8.5.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Main Steam Relief or Safety Valve

Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection system be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

**[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]**

Conclusion

The NRC staff has reviewed the licensee's analyses of the excess heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ~~S~~AFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 14, 15, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2 Decrease in Heat Removal by the Secondary System

2.8.5.2.1 Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)

Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

**[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]**

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2.2 Loss of Nonemergency AC Power to the Station Auxiliaries

Regulatory Evaluation

The loss of nonemergency ac power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

**[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]**

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of nonemergency ac power to station auxiliaries event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ~~S~~AFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of nonemergency ac power to station auxiliaries event.

### 2.8.5.2.3 Loss of Normal Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a LOOP. Loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

**[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]**

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of normal feedwater flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of the loss of normal feedwater flow. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of normal feedwater flow event.

### 2.8.5.3 Decrease in Reactor Coolant System Flow

#### 2.8.5.3.1 Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if SAFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.3.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

**[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]**

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in reactor coolant flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the decrease in reactor coolant flow event.

2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

**[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]**

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition and concludes that the licensee's analyses have adequately accounted for the changes in core design necessary for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 14, 15, and 31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition.

2.8.5.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the AOO and the description of the event itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

**[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]**

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal at power event and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the ~~S~~AFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 14, 15, and 31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal at power.

### 2.8.5.4.3 Startup of a Recirculation Loop at an Incorrect Temperature and Flow Controller Malfunction Causing an Increase in Core Flow Rate

Regulatory Evaluation

A startup of an inactive loop transient may result in either an increased core flow or the introduction of cooler water into the core. This event causes an increase in core reactivity due to decreased moderator temperature and core void fraction. The NRC staff's review covered (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; (3) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and (4) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

**[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]**

Conclusion

The NRC staff has reviewed the licensee's analyses of the increase in core flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ~~S~~AFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 14, 15, 27, 28, and 32 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the increase in core flow event.

### 2.8.5.5 Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

**[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]**

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent operation of ECCS or malfunction that increases reactor coolant inventory and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent operation of ECCS or malfunction that increases reactor coolant inventory.

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of a Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. The pressure relief valve discharges into the suppression pool. Normally there is no reactor trip. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCVs) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

**[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]**

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent opening of a pressure relief valve event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent opening of a pressure relief valve event.

Docket No. 50-271
BVY 04-058

Attachment 5

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 8

Extended Power Uprate

Response to Request for Additional Information

SUMMARY OF COMMITMENTS

# Licensee Identified Commitment Form

This form identifies actions discussed in this letter for which Entergy Nuclear Operations, Inc. (Entergy) commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are not commitments.

| VYNPS COMMITMENT | TYPE (Check one) | | SCHEDULED COMPLETION DATE (If Required) |
|---|---|---|---|
| | One-Time Action | Continuing Compliance | |
| Steam Dryer Inspection During RFO | X | | Fall 2005* |
| Steam Dryer Inspection During RFO | X | | Spring 2007* |
| Steam Dryer Inspection During RFO | X | | Fall 2008* |
| Perform Flow Induced Vibration Monitoring | X | | EPU Implementation |
| | | | |
| | | | |
| | | | |

* CURRENTLY SCHEDULED DATES, SUBJECT TO CHANGE